UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

October 22, 2010

EA-10-087

Mr. Barry Allen
Site Vice President
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2, Mail Stop A-DB-3080
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION SPECIAL INSPECTION TO REVIEW FLAWS IN THE CONTROL ROD DRIVE MECHANISM REACTOR VESSEL CLOSURE HEAD NOZZLE PENETRATIONS 05000346/2010-008(DRS) AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Allen:

On September 9, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed a Special Inspection at your Davis-Besse Nuclear Power Station to evaluate the facts and circumstances surrounding the identification of cracks on March 12, 2010, in the reactor vessel head control rod drive penetration nozzles and J-groove welds. The circumstances associated with this cracking were evaluated against the criteria in Management Directive 8.3, "NRC Incident Investigation Program," and Inspection Manual Chapter 0309, "Reactive Inspection Decision Basis for Reactors." The NRC made the determination that a special inspection would be conducted on March 16, 2010.

The Special Inspection Team reviewed selected procedures and records, observed activities, and interviewed personnel with focus on the areas described in the Special Inspection Charter contained in Attachment 5 of the enclosure of this inspection report. The team confirmed that the nondestructive examinations of the nozzles and J-groove welds met NRC requirements and were successful in identifying cracks at an early stage, such that plant safety was not challenged. The team concluded that your staff had established a strong basis for the direct cause of this cracking, which was primary water stress corrosion cracking (PWSCC). The team also concluded that your staff established an adequate basis for the root cause related to a less than adequate perception of the risk of PWSCC susceptibility with the replacement head. However, the limited evaluation of site staff knowledge and training related to PWSCC was considered a weakness in the root cause determination process. The team confirmed that appropriate nozzles were repaired in accordance with NRC requirements and concluded that the repaired vessel head was suitable to return to service. Further, based on crack growth analyses and the shortened reactor vessel closure head (RVCH) operating period (confirmed in Confirmatory Action Letter (CAL) 3-10-001), the team concluded that margins existed such that the likelihood for PWSCC induced nozzle leakage would remain low for the remaining planned RVCH operating service period. The attached inspection report documents the inspection results, which were discussed with you and your staff at the exit meeting held on September 9, 2010.

Based on the results of this inspection, three NRC-identified findings of very low safety significance were identified. Each finding involved a violation of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your Corrective Action Program, the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy (November 28, 2008). Additionally, during the previous operating cycle, the Davis-Besse Nuclear Power Station was operated with through-wall pressure boundary leakage from cracked control rod drive mechanism (CRDM) nozzles, which was contrary to the Technical Specification 3.4.13 requirement. Operation with pressure boundary leakage of this magnitude (below detection thresholds) would normally be considered a Severity Level IV violation. However, the staff has reviewed your root cause analysis of the event and has concluded that this equipment failure could not have been reasonably avoided or detected by your quality assurance program or other related control measures. Therefore, after consultation with the Regional Administrator, Region III, and the Director, Office of Enforcement, I have been authorized in accordance with Section VII.B.6 of the Enforcement Policy (November 28, 2008), to exercise enforcement discretion and not issue a violation for this issue.

If you contest the subject or severity of the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Davis-Besse Nuclear Power Station. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at Davis-Besse Nuclear Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/
Anne T. Boland, Director
Division of Reactor Safety

Docket Nos. 50-346
License Nos. NPF-3

Enclosure: Inspection Report 05000346/2010-008(DRS);
w/Attachments:
1. Supplemental Information
2. Chronology of Examinations
3. Comparison of Examination Results
4. Pictures- Examinations, Nozzle Repairs, and Flow Paths
5. Davis-Besse Special Inspection Charter

cc w/encl: Distribution via ListServ

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	05000346/2010-008
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Davis-Besse Nuclear Power Station
Location:	Oak Harbor, OH
Dates:	March 16, 2010 through September 9, 2010
Inspectors:	M. Holmberg, Team Lead T. Bilik, Reactor Inspector J. Jandovitz, Project Engineer J. Rutkowski, Senior Resident Inspector, Davis-Besse A. Shaikh, Reactor Inspector A. Wilson, Reactor Inspector J. Collins, Senior Materials Engineer – Division of Component Integrity, Office of Nuclear Reactor Regulation C. Nove, Materials Engineer – Division of Component Integrity, Office of Nuclear Reactor Regulation
Approved by:	D. Hills, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

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SUMMARY OF FINDINGS

Inspection Report (IR) 05000346/2010-008(DRS); Davis-Besse Nuclear Power Station; Special Inspection to Review Flaws in the Control Rod Drive Mechanism Vessel Closure Head Nozzle Penetrations.

On March 12, 2010, with the plant in a scheduled refueling outage, the licensee-identified control rod drive mechanism (CRDM) penetration nozzles, this did not meet acceptance criteria following ultrasonic examinations.

As a result of ultrasonic (UT) examinations, the licensee identified 12 CRDM nozzles with primary water stress corrosion cracking (PWSCC) indications.

As a result of bare metal visual (BMV) examinations, the licensee-identified 14 CRDM nozzle locations with adherent boric acid deposits potentially indicative of leakage. Of these, only CRDM nozzle No. 4 was identified with "popcorn" shaped deposits characteristic of "active leakage."

As a result of dye penetrant (PT) and eddy current examinations on the wetted surface of CRDM Nozzle J-groove welds, the licensee-identified 12 J-groove weld locations with PWSCC indications.

The team confirmed that the UT, BMV, and PT examinations completed during the 2010 refueling outage (RFO) of the CRDM nozzles and J-groove welds met NRC requirements. Additionally, based on review of past reactor vessel closure head (RVCH) examination records, the team concluded that the licensee had adequately completed all previous examinations required by the NRC for the RVCH and CRDM nozzles.

The licensee determined that pressure boundary leakage occurred at CRDM nozzles Nos. 4 and 67 based on UT examination results. The team concluded that the licensee may have been over-reliant on the UT leakage path backwall pattern method used to identify nozzles with through-wall leakage, which could lead to an underestimate in the number of leaking nozzles. This conclusion did not represent a safety concern, because all nozzles with crack indications that could cause leakage were repaired prior to returning them to service.

The team concluded that the licensee established a strong basis for the direct cause of this cracking, which was PWSCC that resulted in flaws propagating through the CRDM nozzle and J-groove welds. The causal factors responsible for the early onset of this PWSCC were a higher RVCH operating temperature than previously assumed, and a random carbide distribution in the Alloy 600 CRDM nozzle material that resulted in a microstructure more susceptible to PWSCC.

The licensee identified that the root cause of this event was a less than adequate perception of the risk of PWSCC susceptibility with the replacement RVCH resulting in inadequate identification, development, and implementation of interim actions to mitigate degradation prior to replacement with a PWSCC resistant Alloy 690 head. However, the team determined that the licensee's actions were reasonable and ultimately successful. In particular, although cracking resulted in minor CRDM nozzle leakage, the licensee identified the PWSCC in the nozzle and J-groove welds at an early stage through NRC required examinations, such that plant safety was not challenged. The team concluded that the licensee had established an adequate basis for the root cause, but identified a weakness in the root cause determination process associated with the limited evaluation of site staff knowledge and training related to PWSCC.

The licensee repaired 24 CRDM nozzles locations with PWSCC indications in the J-groove weld or nozzle base material. The team observed the CRDM nozzle repairs and confirmed that the nozzles were repaired in accordance with the NRC approved relief request and thus were suitable for return to service.

Based on crack growth analyses and the shortened RVCH operating period, the team concluded that margins existed such that the likelihood for PWSCC induced nozzle leakage would remain low for the remaining RVCH operating service period.

This report covers a 177-day period of Special Inspection by the team of NRC inspectors. Three Green findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. The team identified a Non-Cited Violation (NCV) of 10 CFR Part 50 Appendix B, Criterion IX for the licensee's failure to use a nondestructive examination procedure qualified in accordance with applicable Codes and Standards for detection of flaws in control rod drive nozzle repairs. Specifically, the licensee failed to ensure that Procedure 54-ISI-244-10 "Liquid Penetrant Examination of Reactor Vessel Head Penetrations from the Inside Surface," contained a maximum time limit for application of water-wash. The licensee issued a procedure change to incorporate a maximum time limit of 10 minutes for the water-wash application time and demonstrated that this wash time was acceptable.

  This finding was more than minor because if left uncorrected, the failure to use a qualified procedure could become a more significant safety concern. Absent NRC identification, the licensee would not have controlled the maximum times used to wash the penetrant materials off repair weld surfaces. Excessive wash time could have resulted in failure to detect fabrication flaws such as voids and cracks. Undetected cracks returned to service in the repair welds would place the RVCH at increased risk for through-wall leakage and/or nozzle failure. Therefore, this finding adversely affected the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. The issue was corrected promptly, no cracks were returned to service, and the team answered "no" to the Phase I screening question that asked assuming the worst case degradation would the finding result in exceeding the Technical Specification limit for any reactor coolant system leakage. Therefore, the finding screened as having very low safety significance (Green). This finding had a cross-cutting aspect in the area of Human Performance, Work Practices (IMC 0310 (Item H.4(c)) because the licensee did not provide adequate supervisory and management oversight of work activities including contractors such that nuclear safety was supported. Specifically, the licensee failed to provide an adequate oversight in the review and acceptance of the unqualified vendor Procedure 54-ISI-244-10 (Section 4OA3.5).

- Green. The team identified a NCV of 10 CFR Part 50 Appendix B, Criterion IX for the licensee's failure to perform repair welding on control rod drive mechanism nozzle No. 4 using a qualified weld procedure. Specifically, the licensee failed to ensure that the weld procedure supplement PS0140-002 controlled heat input to less than that demonstrated in the supporting weld procedure qualification. To restore compliance, the licensee completed a new weld coupon, tested the coupon, and documented the results in a new procedure qualification record. The procedure qualification record recorded heat inputs for the weld coupon that bound the heat input used for the weld repairs completed on CRDM nozzle No. 4 and the weld coupon test results demonstrated the weld properties were acceptable.

  This finding was more than minor because if left uncorrected, the failure to use a qualified weld procedure could become a more significant safety concern. Absent NRC identification, the licensee would not have completed a Code qualified weld repair on Control Rod Drive Mechanism nozzle No. 4 prior to returning the reactor vessel closure head to service. The repair weld lacked qualification tests to demonstrate that the mechanical properties (toughness, ductility or strength) were adequate, which could have placed the RVCH at an increased risk for through-wall leakage and/or nozzle failure. Therefore, this finding adversely affected the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. The issue was corrected promptly, the unqualified repair weld was not placed in service, and the team answered "no" to the Phase I screening question that asked assuming the worst case degradation would the finding result in exceeding the Technical Specification limit for any reactor coolant system leakage. Therefore, the finding screened as having very low safety significance (Green). This finding had a cross-cutting aspect in the area of Human Performance, Work Practices per IMC 0310 (Item H.4(c)) because the licensee did not provide adequate supervisory and management oversight of work activities including contractors such that nuclear safety was supported. Specifically, the licensee failed to provide an adequate oversight in the review and acceptance of the unqualified vendor weld procedure supplement (PS) 0140-002 (Section 4OA3.5).

- Green. The team identified a NCV of 10 CFR Part 50 Appendix B, Criterion V for the licensee's failure to provide documented instructions appropriate to the circumstances for the remote visual examination of the final dye penetrant examination completed on repaired nozzle No. 61. Specifically, OI 03-1240857-006 "BWOG CRDM Nozzle Top Down Inspection Tooling Operating Instructions," did not include guidance for control of spacer sizes or camera field of view necessary to ensure that the entire examination surface area was viewed. To correct this issue, the procedure was revised to include additional instructions to ensure complete examination coverage with the remote camera system. Additionally, the licensee repeated the examinations on nozzle No. 61 and nine additional nozzles with incomplete examination coverage.

This finding was more than minor because if left uncorrected, the failure to use an adequate procedure for detecting flaws could become a more significant safety concern. Absent NRC identification, the licensee would not have examined the entire surface of the repaired nozzle No. 61 and nine other nozzles, which could have allowed cracks to go undetected. Undetected cracks returned to service in the repair welds would place the RVCH at increased risk for through-wall leakage and/or nozzle failure. Therefore, this finding adversely affected the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. The issue was corrected promptly, weld cracks were not returned to service, and the team answered "no" to the Phase I screening question that asked assuming the worst case degradation would the finding result in exceeding the Technical Specification limit for any reactor coolant system leakage. Therefore, the finding screened as having very low safety significance (Green). This finding had a cross-cutting aspect in the area of Human Performance, Work Practices per IMC 0310 (Item H.4(c)) because the licensee did not provide adequate supervisory and management oversight of work activities including contractors such that nuclear safety was supported. Specifically, the licensee failed to provide an adequate oversight in that no licensee review was completed for the inadequate vendor Procedure OI 03-1240857-006(4OA3.5).

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

BACKGROUND AND EVENT OVERVIEW

On March 12, 2010, with the plant in a scheduled refueling outage, the licensee-identified control rod drive mechanism (CRDM) penetration nozzles, that did not meet acceptance criteria following ultrasonic (UT) examination. Additionally, bare metal visual (BMV) examinations of the outer surface of the reactor vessel closure head (RVCH) identified boric acid deposits indicative of reactor coolant system (RCS) leakage. On March 16, 2010, the NRC chartered (Attachment 5) a Special Inspection Team (SIT) to assess the circumstances surrounding the identification of flaws in the CRDM nozzle penetrations at Davis-Besse Nuclear Power Station. The SIT developed the examination chronology of the replacement RVCH; interviewed plant personnel; and collected and analyzed factual information and evidence relevant to the flaws identified in the CRDM nozzle penetrations and conducted visual inspections of the reactor vessel head. The inspection was conducted in accordance with the SIT Charter, NRC Inspection Procedure 93812, "Special Inspection," and NRC Management Directive 8.3, "NRC Incident Investigation Program." A public exit meeting was conducted on September 9, 2010.

REPLACEMENT VESSEL HEAD HISTORY

Davis-Besse Nuclear Power Station is a two-loop pressurized water reactor designed by Babcock and Wilcox (B&W). The original Davis-Besse RVCH was replaced in 2002 with an unused head from the cancelled Midland Unit 2 plant. The replacement RVCH was certified with an American Society of Mechanical Engineers (ASME) Code N-Stamp on August 27, 1975. However, from 1975 through 2002 the head was stored in the Midland plant and had not been maintained under a 10 CFR Part 50, Appendix B, quality assurance program. In 2002, to confirm that the reactor vessel head had not deteriorated during the storage period, the RVCH was subjected to a series of baseline nondestructive examinations (NDE). These examinations included visual (VT-1) examination of the head surfaces, dye penetrant (PT) examination of the CRDM nozzle J-groove welds, UT examinations of the nozzles, and eddy current (ET) examination of the inside surface of the CRDM nozzles. On March 27, 2004, power operation commenced at the Davis-Besse Nuclear Power Station with the replaced RVCH.

VESSEL HEAD DESCRIPTION AND CONFIGURATION

The RVCH is constructed from low alloy steel (ASME Code, SA-533, Grade B, Class 1), and has a torispherical shape with approximately an 87-inch inside crown radius and a minimum head thickness of 6.63 inches. The inside surface of the vessel head is clad with Type 308 stainless steel, nominally 3/16 of an inch thick. The cladding is provided for corrosion resistance and is not credited as pressure boundary material. The RVCH was manufactured by B&W as a Class A component in accordance with the ASME Code Section III, 1968 Edition, summer 1968 Addenda.

The RVCH has 69 CRDM penetration nozzles arranged in a rectangular pattern, with a center-to-center distance of approximately 12 inches, and are numbered sequentially starting at the center and progressing concentrically outward. The nozzles are fabricated from Alloy 600 tubes, with an outside diameter of approximately 4.00 inches and a wall thickness of approximately 0.65 inch. The nozzles vary in length, depending on the location on the vessel head, from approximately 30 inches in the center to approximately 50 inches on the periphery, with a minimum of two inches that protrudes below the reactor closure head. The nozzles extend through 4.00 inch bores in the vessel head, and are welded to the head with a J-groove weld at the inner surface of the head fabricated with Inconel Alloy 182 filler material. The top end of the nozzle terminates in a flange that supports the CRDM housings. A service structure is attached to the reactor vessel head, which is approximately 18 feet high and is 10 feet in diameter. The service structure stabilizes the CRDMs and contains a horizontal layer of metallic reflective insulation approximately 2 inches above the top of the vessel head. The CRDM nozzles pass through the insulation layer and attach to the CRDM housings with bolted flanges located about 9 inches above the horizontal insulation layer. Attachment 4, Picture No. 8 illustrates the service structure, CRDM nozzle, and RVCH configuration.

4. OTHER ACTIVITIES (OA)

4OA3 Special Inspection (93812)

.1 Establish the Pertinent Examination Chronology/History of the Replacement RVCH

a. Inspection Scope

The team reviewed licensee examination records for the replacement RVCH related to the J-groove welds and CRDM penetration nozzles from preservice through the current outage inspection to establish the pertinent chronology of examinations.

The team with assistance from the office of Nuclear Reactor Regulation (NRR) staff performed a review of licensee docketed responses to NRC requirements or requests related to inspection of the RVCH. The team compared these responses with the licensee’s inspections completed to date to determine if the licensee had met NRC requirements. Specifically, the team reviewed the licensee’s responses to the NRC Order EA 03-009, “Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors,” NRC Order EA 03-214, “Confirmatory Order Modifying License,” and 10 CFR 50.55a(g)(6)(ii)(D) to determine if the licensee had completed the NRC required inspections of CRDM nozzles and the RVCH.

The team also reviewed the Licensee Event Report (LER) 05000346/2010-002-00 “Control Rod Drive Nozzle Primary Water Stress Corrosion Cracking (PWSCC) and Pressure Boundary Leakage” to evaluate the licensee’s compliance with Technical Specifications (TS) 3.4.13 “Reactor Coolant System (RCS) Operational Leakage” during past operating cycles. In support of this review, the team evaluated the chemical and radiological analysis of boric acid samples collected on the RVCH by the licensee to determine when these leakage deposits were formed. Additionally, the team reviewed data related to detecting RCS unidentified leakage for both Operating Cycle Nos. 15 and 16 to determine if indications/trends for unidentified reactor coolant leakage existed. Specifically, this review included:

- the results from calculated RCS leak rates with emphasis on differences between operating cycles and, for Cycle No. 16, any indications of sustained increases in RCS unidentified leak rate;
- containment airborne activity levels recorded for each cycle; and
- the FLUS system data recorded for both cycles.

b. Findings and Observations

b.1 Chronology of RVCH Examinations

A chronology of the replacement RVCH examinations including the CRDM nozzles and J-groove welds was developed (Attachment 2 to this Report). Based upon of these previous RVCH examination records, the team concluded that the licensee had adequately completed all examinations required by the NRC for the RVCH and CRDM nozzles. The licensee also completed additional voluntary examinations of the RVCH, CRDM nozzles, and J-groove welds not required by NRC regulations.

b.2. RVCH and CRDM Nozzle Inspection Requirements

Implementation of Current Head Inspection Requirements

Title 10 of the Code of Federal Regulations, 50.55a “Codes and Standards” Paragraph g(6)(ii)(D), contains the inspection requirements for CRDM nozzles and RVCH at pressurized water reactors (PWR). These requirements were established on September 10, 2008, and superseded the previous NRC requirements for RVCH and CRDM nozzle inspections established under NRC Order EA 03-009. This revised regulation required the licensee to implement the ASME Code Case (CC) N-729-1 “Alternative Examination Requirements for PWR Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1” with conditions. The CC N-729-1 inspection requirements were similar to the previous inspection requirements identified in NRC Order EA 03-009 with a notable exception. Item B4.20 of Table 1 of CC N-729-1 introduced a reinspection year (RIY) term that defined a limit of 2.25 for the maximum operating time between volumetric examinations. The RIY limit was derived from a calculation based on operating head temperature and the effective full power years of operation accumulated during the operating period. The licensee determined that the RIY limit was applicable to the operating cycle following the first inservice volumetric examination of the replacement RVCH. The replacement RVCH was first placed in service to support power operations in 2004 and the 2010 refueling outage UT examination of the CRDM nozzles represented the first inservice volumetric examination. Therefore, the licensee applied the RIY limit of 2.25 to the next Davis-Besse Operating Cycle (No. 17).

The effective date of the revision to 10 CFR 50.55a(g)(6)(ii)(D), which first required the licensee to implement CC N-729-1, was October 10, 2008. The team reviewed 10 CFR 50.55a(g)(6)(ii)(D), including the statement of considerations to determine if the licensee had correctly implemented CC N-729-1. The licensee’s 2010 inservice volumetric examination of the RVCH occurred within eight calendar years of the baseline volumetric examination completed prior to placing the RVCH in service, which met one of the two conditions associated with the maximum time allowed between volumetric examinations identified in Item B4.20 of Table 1, of CC N-729-1. The NRC concluded that the licensee’s decision to apply the RIY limit after the first inservice volumetric examination was reasonable because the licensee had met the second condition (eight calendar years) associated with the maximum time between volumetric examinations.

Late Update of ISI Program with Head Inspection Requirements

Title 10 of the Code of Federal Regulations, 50.55a "Codes and Standards" Paragraph g(6)(ii)(D) requires licensees to implement the ASME CC N-729-1 with conditions. One condition of this regulation directed the licensee to augment their inservice inspection (ISI) program by December 31, 2008, to incorporate the requirements of CC N-729-1. However, the licensee identified that the ISI Program had not been updated to incorporate the CC N-729-1 requirements until February 24, 2010. Because this error did not result in failure to complete the required RVCH and CRDM nozzle examinations this violation was of minor significance. The licensee entered this violation of 10 CFR 50.55a(g)(6)(ii)(D) into the corrective action system (CR 10-74253).

Implementation of Previous Head Inspection Requirements

The team evaluated the licensee's compliance with the previous head inspection requirements established under NRC Order EA 03-009 revised on February 20, 2004, and NRC Order EA 03-214 issued on March 8, 2004. In accordance with NRC Order EA 03-214 the licensee was required to perform a BMV examination of the head during the mid-cycle outage for the No. 14 operating cycle in January of 2005. The licensee completed this examination with no evidence of leakage. In accordance with NRC Order EA 03-009, for the replacement head, the licensee was required to complete a BMV examination every third refueling outage or 5 years, whichever occurs first. Additionally, the licensee was required to perform a volumetric examination at least every 4 refueling outages or seven years, whichever occurs firsts. By refueling outage (RFO)-15, which started on December 30, 2007, (2nd RFO for the replaced head and last outage governed by the requirements of NRC Order EA 03-009), the head had accumulated 6.28 Effective Degradation Years (EDY) and thus only required a BMV examination. The 6.28 EDY value was obtained from calculation C-ME-099.99-013 "EDY Calculation for Alloy 600/82/182 PWSCC Susceptibility Determination," which utilized a revised higher bounding head operating temperature of 615 degrees Fahrenheit (^{o}F). The licensee had completed a BMV examination of the head that met the requirements of NRC Order EA 03-009 during each RFO (RFO-14 in 2006 and RFO-15 in 2008) without evidence of CRDM nozzle leakage. Therefore, the team concluded that despite the head operating at a higher temperature of 615 ^{o}F than previously known (Section 4OA3.3); the licensee had met the head inspection requirements of NRC Order EA 03-009.

b.3 (Closed) Licensee Event Report (LER) 05000346/2010-002-00: Control Rod Drive Nozzle PWSCC and Pressure Boundary Leakage

On March 12, 2010, during UT of CRDM nozzles, the licensee identified nozzles, which did not meet acceptance criteria. Additionally, during BMV examination of the outer surface of the RVCH, the licensee identified boric acid deposits indicative of RCS leakage. The direct cause of this event was PWSCC of the CRDM nozzles and J-groove welds and the license identified and repaired a total of 24 CRDM nozzles with PWSCC in the nozzle or J-groove welds. Because the PWSCCs identified were well below crack sizes required for nozzle ejection and there was no discernable head wastage, the licensee concluded that this issue was of very low safety significance. The team evaluated the safety significance of this cracking (Section 4OA3.4) and concluded that the cracking was identified early such that plant safety was not challenged.

This event was reviewed in-depth by the team as discussed within each section of this report. The NRC determined that past operation with CRDM nozzle leakage was a violation of TS 3.4.13 "RCS Operational Leakage" and applied enforcement discretion to not issue a violation for this issue as discussed in the following report section. Documents reviewed as part of this inspection are listed in Attachment 1. LER 05000346/2010-002-00 is closed.

b.4 Technical Specification (TS) Leakage Requirements

Introduction: A self-revealed violation of TS 3.4.13 "RCS Operational Leakage" was identified associated with pressure boundary leakage through cracked CRDM penetration nozzles during the prior operating cycle. Because the licensee appropriately implemented their quality control program, and this violation was the result of unavoidable equipment failure, the NRC has elected to exercise enforcement discretion and not issue a violation.

Description: The team evaluated the licensee's compliance with TS 3.4.13 "RCS Operational Leakage" during past operating cycles.

The UT examination of the CRDM nozzles identified crack indications in 12 CRDM nozzles. The licensee concluded that only CRDM nozzles Nos. 4 and 67 experienced through-wall leakage (Section 4OA3.b4). Based on identification of boric acid deposits near nozzle penetrations on the RVCH, and isotopic analysis of these boric acid deposits, the licensee concluded that leakage had occurred through the CRDM nozzles during the previous Operating Cycle (No. 16). With evidence of leakage during the last operating cycle, the licensee concluded that they operated the plant in a condition prohibited by TS 3.4.13 "RCS Operational Leakage" and reported this condition to the NRC in LER 05000346/2010-02-00. The licensee staff stated that, although the plant was operated in a condition prohibited by TS, they did not violate TS 3.4.13 because the leakage rate was below that which could be identified by the TS surveillance requirement of 3.4.13.1 using an RCS water inventory balance. The NRC reviewed this position and did not agree with the licensee's conclusion on TS 3.4.13 compliance as discussed in the Enforcement Section below.

Analysis: The team reviewed the root cause analysis of the event and RCS leakage data from previous operating cycles, and concluded that the equipment failure (cracked CRDM nozzles) could not have been avoided or detected by the licensee's quality assurance program or other related control measures. Therefore, the team concluded that a licensee performance deficiency did not exist for this violation and did not apply to the Significance Determination Process as described in IMC 0609. Although, a quantitative risk-evaluation was not completed, the licensee performed a deterministic evaluation of the safety significance for the PWSCC identified in the CRDM nozzles (Reference LER 05000346/2010-002-00). Because the PWSCC sizes identified in the CRDM nozzles were well below the crack size that would challenge structural integrity and there was no discernable head wastage, the licensee concluded that this issue was of very low safety significance. The team agreed with the licensee's assessment that this issue was of very low safety significance.

Enforcement: Technical Specification 3.4.13 required that RCS operational leakage shall be limited to "No pressure boundary LEAKAGE" when in Modes 1 through 4. Contrary to this requirement, during Operating Cycle No. 16, which ended on February 28, 2010, the licensee operated the Davis-Besse Nuclear Power Station in Mode 1 with pressure boundary leakage from cracked CRDM nozzles Nos. 4 and 67.

Operation with pressure boundary leakage of this magnitude (below detection thresholds) would normally be considered a Severity Level IV violation. The NRC may exercise discretion and refrain from issuing a civil penalty and/or a Notice of Violation after considering the general principles of the Enforcement Policy (November 28, 2008) and the surrounding circumstances. Because the licensee met all associated NRC regulations with regard to CRDM nozzle inspections and the violation was the result of equipment failure that could not have been reasonably avoided or detected, the NRC elected to apply Section VII.B.6 of the Enforcement Policy (November 28, 2008), and exercise enforcement discretion to not issue a violation. Specifically, had the licensee used a more accurate estimate of head operating temperature, earlier volumetric examinations would still not have been required under NRC requirements that existed at that time (Section 4OA3.1.b.2). In addition, the licensee did not miss any available indicators of leakage such that they could have identified it earlier (Section 4OA3.1.b.6).

b.5 Boric Acid Sample Results Indicate Leakage from Previous Operating Cycle

During the 2010 refueling outage Cycle No. 16, the licensee collected three samples of boric acid deposits from the top of the RVCH. One sample was collected at the head-to-nozzle interface of nozzle Cycle No. 4. The second sample was collected from the east side of the RVCH (opposite side from CRDM nozzle No. 4). The third sample was collected from deposits accumulated along the inside wall of the service structure. The licensee performed an isotopic analysis (gamma spectrometry) and elemental analysis (inductive-coupled plasma mass spectroscopy) on these boric acid samples.

The boric acid deposit analyses results were documented in a vendor Report 51-9136423-000 "Analysis of Davis-Besse Boric Acid Deposits from RFO-16 CRDM Nozzle Inspections." This report identified the following results:

- Leakage occurred during periods of operation (or startup or shutdown periods during which lithium was present). Significant lithium was present in the samples relative to boron.
- The samples had evidence of thermal conditioning. Specifically, the boron-to-lithium ratio in all three samples was less than that present in the reactor coolant, suggesting that lithium borate compounds formed with volatilization of some of the excess boric acid in the residue from heating.
- The residue samples from nozzle No. 4 and the East Side of the RVCH had characteristics of leakage during the last month of Cycle No. 16 operation. Short-lived isotopes CS-136 and I-131 could only be present from leakage that occurred during this period, although earlier leakage could also be present.
- The residue from other locations on the RVCH had an earlier leakage history, possibly occurring during or after recovery from the high pressure turbine bearing oil leak power reduction in December 2008 and subsequent operations, or recovery from the pressurizer Code safety valve outage in April through May of

2009. The CS-134 to CS-137 isotopic ratio matches the RCS ratio: (1) during the recovery period from the turbine oil leak power reduction; (2) recovery from the pressurizer Code safety valve outage in April-May of 2009; and (3) during the power reduction for RFO-16 on February 27, 2010. Since short-lived isotopes were not identified, leakage during the power reduction for RFO-16 can be ruled out. The isotopic CS-134 to CS-137 ratio was also lower than that for the other samples, which indicated a longer history.

- The current samples indicate that much less corrosion had occurred from exposure to boric acid relative to the 2002 samples collected from the original RVCH prior to replacement. The iron, nickel, and chromium levels relative to boric acid were much lower, indicating less corrosion of carbon steel, stainless steel, or weld metals.

Following Operating Cycle No. 15, no evidence of leakage (e.g., boric acid deposits) was identified on the RVCH during the BMV examination. Therefore, the licensee evaluated the boric acid analysis results in conjunction with the BMV examination results and concluded that pressure boundary leakage had occurred during Operating Cycle No. 16 (December 2008 through February 2010).

b.6 <u>Operation Leakage History- Reviews of Leakage Indications and Trends</u>

The team reviewed data related to identified and unidentified RCS operational leakage collected by the licensee during the previous two operating cycles. Based on this review as discussed below, the team did not identify any indications or trends in RCS unidentified leakage that could be attributable to reactor head leakage. Unexpected changes in unidentified leakage during Cycle No. 16 that were investigated by the licensee were primarily attributable to measurement data scatter, changes in interfacing system boundary valve leakage, or valve packing leakage. Containment atmosphere radionuclide levels appeared consistent with the previous operating cycle although somewhat higher concentrations existed during Cycle No. 16 due to a higher number of degraded fuel pins in this operating cycle. Additionally, the FLUS system, which only monitors for leakage from the lower portion of the reactor vessel, did not detect any RCS leakage.

The primary method used by the licensee for determining RCS leakage is an inventory balance. This method requires determining the mass change in RCS and makeup system water inventories typically over a 4-hour period of steady-state power operation. The team used the licensee's results from these inventory balance calculations and determined the average and standard deviations for the unidentified leakage rates for each operating cycle. For operating Cycle No. 15 the unidentified leak rate mean and standard deviation were 0.006 gallons per minute (gpm) and 0.031 gpm respectively. For operating Cycle No. 16 the unidentified leak rate mean and standard deviation were -0.008 gpm and 0.021 gpm respectively. These results were well below the licensee's estimated accuracy for the inventory balance method. Specifically, the licensee determined that the accuracy of this inventory balance was +/-0.140 gpm and leakage rates below 0.050 gpm are difficult to detect. Based on evaluation of the licensee's information with emphasis on unidentified leakage, the team did not identify substantive difference in Cycle No. 16 as compared to Cycle No. 15, which could be attributable to reactor vessel upper head leakage.

Boric acid leakage and corrosion products introduced into the containment atmosphere can be identified by evaluating the deposits which buildup on the containment cooler tubes, or on the containment radiation monitoring system filters (Reference NRC IR 05000346/2002002003). During post-shutdown inspections of containment air cooler tubes the licensee did not identify boric acid deposits, which suggested that borated water vapor was not present in the containment. Additionally, during Cycle No.16 operation of the continuous containment airborne radioactivity monitors, the licensee stated that no clogged filter media was identified that would be indicative of corrosion particles or boric acid deposits.

The licensee recorded data collected from the continuous containment airborne radioactivity monitors for the previous two operating cycles. The team noted that the containment gaseous activity in Cycle No. 16 ranged from about 3 to 10 times higher than in cycle No. 15. The containment airborne iodine and particulate activity cycle for Cycle No.16 was only slightly higher (approximately 30 percent higher; a factor of about 1.35) than in Cycle No. 15. Additionally, for the last three months of Cycle No. 16, the containment particulate activity increased from a nominal value of 2.2E-10 micro-curies per cubic centimeter (uCi/cc) to approximately 4.1E-10 uCi/cc and in the weeks just before the end-of-cycle No. 16 this value decreased to about 3.3E-10 uCi/cc. The licensee attributed these anomalies to the higher RCS gaseous activity levels in Cycle No. 16 caused by an increase in the numbers of degraded fuel pins in Cycle No. 16 relative to Cycle No. 15. The team considered this a reasonable explanation for these cycle-to-cycle variations in gaseous and particulate containment airborne activity levels.

In 2003, the licensee installed an under reactor vessel humidity monitoring system (FLUS) to help detect any potential leakage from the incore instrumentation (ICI) penetrations located on the lower vessel head. The ICI penetrations are constructed of similar materials to those used in the upper head CRDM nozzle penetrations, but typically operate at a lower temperature. The FLUS system is designed to monitor only the lower portion of the reactor vessel and leakage from the upper head CRDM penetration nozzles would not be detected by this system. The minimum threshold for sensitivity of the FLUS system ranges from of 0.05 gpm to 0.005 gpm and the licensee stated that the installed system has a sensitivity of about 0.01 gpm. To ensure that unidentified RCS leakage did not exist from the ICI penetrations, the team reviewed data collected from this system. Specifically, the team reviewed humidity levels recorded from under the enclosed space of the reactor vessel and insulation and compared this data to humidity measurements outside of this space. The team did not identify any significant difference between these readings from either cycle, which indicated that RCS leakage did not occur from the lower vessel head area.

.2 <u>Compare Current Examination Results with Samples of the 2005 – 2008 Examination Records and Pre-Service Records to Determine Whether the Conditions Were Pre-Existing.</u>

a. <u>Inspection Scope</u>

The team performed an on-site review and assessment of the 2005 – 2008 examination records and pre-service records to determine whether the conditions were pre-existing. Specifically, the team compared the current examination with prior examination records on an individual nozzle basis for any nozzle identified during the current refueling outage with:

- Boric acid deposits potentially indicative of leakage;
- PWSCC indications identified by UT examinations; and
- Rejectable indications in the J-groove weld surface identified by PT and ET examinations.

For the CRDM nozzles with PWSCC indications identified during UT examinations and other nozzles with suspected UT anomalies, the team performed a review of the current and preservice UT data recorded on electronic files. The team made use of the licensee's vendor software and data analysis systems to compare data recorded from similar UT search units at locations where potential flaw indications were observed in the 2010 data files. The team completed this review to independently determine if the UT flaw indications were present in the preservice data.

b. Findings and Observations

The team concluded that the flaws identified during the current outage were not pre-existing, with the exception of CRDM nozzle Nos. 4 and 64. For nozzle No. 4, indication No. 2 identified during the 2010 UT examination could have propagated from an original construction weld indication identified during the 2002 preservice UT examination. At CRDM nozzle No. 64, a J-groove weld indication was identified during a PT examination at the same location as an original fabrication weld repair. Therefore, this flaw could possibly have propagated from this original fabrication weld repair. The team reviewed the RVCH construction NDE records and confirmed that both nozzle Nos. 4 and 64 had met the construction Code acceptance criteria prior to placing the head in service. The team's comparison of NDE results is recorded in Attachment 3 to this report.

.3 Evaluate The Adequacy of the Licensee's Plan for Assessing the Causes of the Flaws and the Licensee's Rationale Regarding Acceptability of the Head for Continued Service.

a. Inspection Scope

The team performed an on-site review and assessment of the scope, charter, and composition of the licensee's Root Cause Team (RCT) to assess the effectiveness of the RCT in identification of the causes for the CRDM nozzle and weld cracking.

The team reviewed the Root Cause Analysis Report for this event approved by the licensee on May 26, 2010, and documented in Condition Report 10-73323. Specifically, the team reviewed the causes and licensee CA plans in this report to determine if the scope of causes and the corrective actions assigned would preclude recurrence of a similar event.

Additionally, the team reviewed the licensee's analysis of the head operating service life, to determine if the RVCH would continue to operate with a low likelihood for PWSCC induced nozzle leakage.

b. Findings and Observations

Based upon review of the licensee's RCT investigation and root cause report, the team concluded:

- Composition and staffing of the licensee's RCT and RCT Charter were appropriate to identify the cause for the CRDM nozzle flaws and the RCT made appropriate decisions on the gathering of field data to support the root cause investigation.
- A strong basis existed to support the direct cause of this event, which was PWSCC that resulted in flaws propagating through the CRDM nozzle and J-groove welds.
- Corrective actions for the direct cause were sufficient to preclude recurrence with consideration for uncertainties, because of the margins that exist to the onset of PWSCC induced leakage.
- The licensee established an adequate basis for the root cause related to a less than adequate perception of the risk of PWSCC susceptibility with the RVCH. However, the limited evaluation of site staff knowledge and training related to PWSCC was considered a weakness in the root cause determination process.

Based on crack growth analyses and the shortened RVCH operating period, the team concluded that margins existed such that the likelihood for PWSCC induced nozzle leakage would remain low for the remaining RVCH operating service period.

The licensee-identified that the causal factors responsible for the early onset of this PWSCC were, a higher RVCH operating temperature than previously assumed, and a random carbide distribution in the Alloy 600 CRDM nozzle material that resulted in a microstructure more susceptible to PWSCC.

b.1 Licensee Root Cause Team Staffing and Charter

The licensee initially formed an 11 member RCT that was augmented to include 16 members by May of 2010, to determine the cause of the service induced flaws identified in the nozzles and J-groove welds. The licensee's RCT was initially led by the Beaver Valley Power Station Manager for Technical Services and then transitioned to the Davis-Besse Plant Engineering Manager. The RCT staff members included; Senior Engineers from Davis-Besse, Beaver Valley, the Electric Power Research Institute (EPRI), and AREVA. The RCT also included a Materials Engineer from Westinghouse, who was also the Chairman of the PWR Owner Group Materials Subcommittee - Materials. The NRC team considered the licensee's RCT member composition and experience appropriate to complete the RCT Charter.

To confirm the direct cause of the cracking, the licensee RCT elected to harvest boat and ring material samples from nozzles with crack indications for metallurgical examination and destructive testing. The RCT applied a systematic method for making decisions on which CRDM locations with crack indications would provide the best data for metallurgical analysis (CRDM nozzles Nos. 4 and 10 were selected). The RCT Charter also included evaluation of the potential "crack" indications to determine if the initiation time was consistent with industry correlations. Because crack growth-rate testing of the nozzle material samples would take an extended period of time (several months); the RCT evaluation relied on correlating field NDE results with industry data on crack growth-rates. Additionally, the licensee RCT directed the collection of boric acid samples from the RVCH for chemical and isotopic analysis to determine when these deposits were formed. The NRC team concluded that the RCT made appropriate decisions on the gathering of field samples and data to support the root cause investigation and report conclusions.

As of May 26, 2010, the licensee's RCT identified six potential failure modes, with failure Mode 1 subdivided into seven topics. Of these, the direct cause for this leakage was determined to be Failure Mode 1- PWSCC. This failure Mode 1 considered seven sub-topics; tube production, fabrication, welding, transportation, primary system chemistry, core streaming (e.g., uneven mixing of fluid temperature in the RVCH), susceptible material, environment and stress. The failure modes considered by the RCT, but ruled out, included; lack of fusion, baseline inservice inspection, thermal fatigue cracking, mechanical fatigue cracking, and environmental fatigue cracking. The NRC team concluded the licensee's direct/physical cause was consistent with the observed condition of the bare metal head, preliminary metallurgical results and UT results with consideration for materials, operating history and environment.

b.2. <u>Root Cause Report Conclusions</u>

In Condition Report 10-73323, the licensee documented a root cause analysis report for the CRDM nozzle and weld cracking with pressure boundary leakage. The licensee identified two causes for the CRDM nozzle cracking and J-groove weld flaws with pressure boundary leakage:

- The direct cause was PWSCC that resulted in flaws propagating through the CRDM nozzle or through the length of the J-groove weld and boric acid migrating onto the RVCH.
- The root cause was a less than adequate perception of the risk of PWSCC susceptibility with the replacement RVCH resulting in inadequate identification, development, and implementation of interim actions to mitigate degradation prior to replacement with a PWSCC resistant Alloy 690 head.

The licensee identified three contributing causes for the CRDM nozzle cracking and J-groove weld flaws with pressure boundary leakage:

- A contributing cause was a less than adequate understanding of risk that also led to missed opportunities by the organization to recognize, more aggressively interrogate, and then validate actual RVCH operating temperatures and to correlate those temperatures to potential premature PWSCC development.
- A second contributing cause was a less than adequate understanding of risk that also led to an organizational belief that meeting Code and Confirmatory Order required inspections of the RVCH was sufficient to prevent pressure boundary leakage for the replacement RVCH versus a more conservative approach.
- A third contributing cause was a less than adequate understanding of risk that also led to less than adequate management oversight and involvement in the ISI program related to RVCH nondestructive examinations, and the understanding of

ongoing industry developments related to Alloy 600 PWSCC and the associated changes in regulatory requirements.

b.3. Licensee Corrective Actions

As of May 26, 2010, in Condition Report 10-73323, the licensee identified 22 corrective actions for the CRDM nozzle and J-groove weld cracks. For example:

- Modification and repair of 24 CRDM nozzles;
- Inspection of the modified and non-modified CRDM nozzles;
- Tracking the installation of a new head made of PWSCC resistant materials;
- PT or ET examinations of nozzle J-groove welds during future outages that were not removed from service;
- Assignment of repetitive tasks to monitor and calculate head temperature and RIY limit every three months;
- Design fuel Cycle No. 18 core reload to reduce peak temperatures in the 21 central fuel assemblies;
- Lessons-learned training for engineering staff on the 16 RFO head root cause; and
- The licensee also proposed additional actions for this issue which included an evaluation to determine if an Operating Experience Report should be issued and evaluation for a functional or condition monitoring type failure and an effectiveness review. The licensee's Effectiveness Review Plan to assess the effectiveness of the root cause corrective actions included a survey or interview of 10 percent of engineering and supervisor personnel after completion of lessons learned and case study training on this issue.

b.4 Assessment of Direct Causes and Contributing Factors

b.4.1 Direct Cause of Leakage - PWSCC

The licensee RCT identified that the direct cause of this event was PWSCC that resulted in flaws propagating through the CRDM nozzle or J-groove welds. To confirm the direct cause of the CRDM nozzle leakage, the licensee removed boat samples of material from nozzles Nos. 4 and 10. The licensee contracted with a vendor to perform examinations and tests on these removed samples which included; PT examinations, scanning electron microscope (SEM) surface examinations, SEM-energy dispersive spectroscopy, micro-focused X-ray inspections, metallographic examinations, knoop micro-hardness measurements and fractographic examinations of two cracks. The results of these examinations and tests were documented in a Westinghouse report RTU-MCE-10-36, "Final Report-Summary of Davis-Besse Unit 1 CRDM Nozzle Boat Samples Destructive Examinations." This report confirmed that PWSCC likely initiated from the outside diameter surface of the nozzle at or near the heat affected zone. The team concluded that these physical tests/examinations of nozzle materials established a strong basis to support the licensee's conclusion that PWSCC caused the flaws in the nozzles and J-groove welds and the RVCH leakage.

The licensee assigned two corrective actions credited with precluding recurrence of the direct cause: PWSCC that resulted in flaws propagating through the CRDM nozzle or through the length of the J-groove weld and boric acid migrating onto the RVCH.

Corrective Action 10-73323-CA13, required the site staff to track completion the installation of a new head made of PWSCC resistant materials, and Corrective Action 10–73323-CA12, required additional surface NDE of the J-groove welds during RFO 17. This second corrective action was superseded by a licensee commitment to shutdown the Davis-Besse plant by October 1, 2011, and replace the RVCH with one fabricated from PWSCC resistant materials. The team concluded that these corrective actions were sufficient to avoid recurrence of PWSCC induced leakage because of the operating margins established by the licensee as discussed in Section 4OA3.3.b.6.

b.4.2 Contributing Factors for PWSCC – RVCH Operation Temperature and Nozzle Material

The factors that affect crack initiation and growth for CRDM penetration nozzles susceptible to PWSCC were previously reviewed and documented by the NRC (reference NRC inspection report (IR) 05000346/2002003(DRS)). The susceptibility of a CRDM nozzle to PWSCC is dependent on material, operating temperature, time, environment and residual stress. Because the operating environment of domestic pressurized water reactors is similar, the susceptibility of a particular nozzle to cracking may be dependent upon time, temperature, material microstructure, and residual tensile stress. Thus, a particular heat of Alloy 600 used to fabricate a penetration nozzle may be more likely to experience cracking as each of these variables is increased (e.g., longer service time, higher operating temperatures, or a higher residual tensile stress). In particular, the initiation and growth of PWSCC flaws is strongly dependent on temperature. For this reason, the RIY limit (CC N-729-1) for maximum operating time between volumetric examinations of CRDM nozzles is based on a calculation that relies on RVCH operating temperature and time in service.

The RCT identified that RVCH operating temperature and random carbides present in the nozzle material were significant contributors to the relatively early onset of PWSCC at Davis-Besse. The team agreed with this conclusion, but identified potential sources of uncertainties with respect to the factors that may have contributed to the relatively early onset of PWSCC as discussed below:

- The licensee concluded that potentially adverse reactor coolant chemistry species (e.g., sulfates) had a low impact on RVCH PWSCC because the RCS chemistry was within industry guidelines for the previous operating periods. However, this basis was weak given the presence of sulfates identified on a discrete particle removed from a crack face during destructive examination of PWSCC material removed from nozzle No. 4.
- The licensee did not provide a documented evaluation for the impact of storage and preservice chemistry conditions (e.g., dry and wet layup) of the replacement Midland head prior to commencing power operation at the Davis-Besse site in March of 2004. The length of time the Davis-Besse replacement RVCH remained in layup conditions is much longer and therefore unique when compared to other B&W heads that have been placed in service.
- The licensee did not quantify the uncertainty in development of the flow model used to estimate the RVCH operating temperature.

- The licensee did not provide a substantive explanation for the longer times predicted for the onset of leakage using 75 percentile crack growth-rate values compared with the actual time to leakage at nozzle No. 4 (less than 5.5 effective full power years (EFPY)).

These uncertainties potentially affected the licensee's evaluation of the head operating service life, before PWSCC induced leakage could recur. To bound these uncertainties, the licensee elected to reduce the operating service period of the repaired RVCH as discussed in (4OA3.3.b.6).

RVCH Operating Temperature

The licensee's previous "best estimated" RVCH operating temperature was based on the average of the narrow range hot leg recirculation loop temperature instruments. The RCT identified that RVCH operating temperature was higher than previously known and was significant contributor to the relatively early onset of PWSCC at Davis-Besse. The RCT conclusions were based in part, on the analysis of head vent line temperature data recorded during the previous operating cycle. A continuous head vent line is installed at a spare head penetration for venting of non-condensable gases. This line allows 0.5 percent of the total core flow to exit the RVCH and return to the RCS. This vent line is instrumented with Type K thermocouple temperature elements that feed two computer data points (T012 and T013) that are monitored and recorded. Because these instruments are not calibrated, the licensee performed a calibration and calculation C-ICE- 062.01-001 to estimate the actual temperature for the continuous head vent line. The licensee concluded that computer points T012 and T013 had -2.6 ^{o}F and -0.7 ^{o}F degree errors respectively and vent line temperature had reached a maximum of approximately 615.4 ^{o}F during operating cycle No. 16. Therefore, portions of the RVCH near the vent line may have operated nine degrees higher than the previously assumed (606.4 ^{o}F) hot leg temperature.

The licensee subsequently determined that channeling of water to the head directly from fuel assemblies located in core locations below the control rod guide tubes was responsible for the increased RVCH operating temperature. Specifically, for the Davis-Besse reactor design, the reactor coolant rises up through the reactor core to remove heat generated by the fuel assemblies. A plenum assembly with control rod guide tubes is located between the reactor core and the RVCH. The plenum assembly is designed to promote a more uniform radial pattern for reactor coolant exiting towards the RCS outlet nozzles. A portion of the total RCS flow proceeds up the control rod guide tube and exits directly into the upper RVCH region (Attachment 4, Picture No. 10). This flow then exits near the periphery of the plenum assembly. Because of this effect, the licensee developed a methodology for better estimating head operating temperature as documented in 51-9137401–000 "Evaluation of Fluid Temperature in DB RV Closure Head." This document estimated fluid temperatures within the RVCH region based on flow distributions, fluid temperatures from other analyses, operating data, and core design analyses. The RVCH head fluid temperatures derived from this calculation were higher than previously calculated due to the non-uniform mixing (channeling) of water exiting fuel assemblies in the core. Because document 51-9137401–000 relied extensively on engineering judgment to estimate internal vessel flowrates as inputs in calculation of fluid temperatures within the head, it was subject to an undefined level of uncertainty. To bound the uncertainty in using this method to derive a "best estimate" of the RVCH operating temperature, the licensee elected to use the highest temperature fuel assembly located in the center group of assemblies (over a given period of operation).

With the RVCH operating temperature above 609°F, the RIY limit in CC N-729-1 requires the licensee to perform volumetric examinations of the CRDM nozzles at frequency of less than two years. Because the volumetric examination must be done with the reactor shutdown, operation of the RVCH is limited to less than a full operating cycle. This condition is unique to Davis-Besse, and differs from other US PWRs. Specifically, other US PWRs operate with a lower head temperature and consequently perform volumetric inspections during scheduled refueling outages before reaching the 2.25 RIY limit. Therefore at other PWR plants, additional operating margins exist to account for the uncertainty of the head operating temperature calculations or measurements. However, this margin is no longer applicable to the Davis-Besse plant, so the use of an accurate or conservative head temperature is essential to ensure an effective inspection program. Because the magnitude of uncertainty had not been quantified in the AREVA 51-9137401–000 document, the licensee elected to use the highest temperature fuel assembly from the centermost 21 assemblies. Although, the licensee believed that this was a conservative or bounding estimate for head temperature, the magnitude of error in calculating fuel assembly temperatures was also not quantified. Therefore, the team requested the licensee quantify the uncertainty in head temperature to support an evaluation (4OA3.3.b.6) for continued operation. Instead, the licensee elected to replace their head during an outage to begin on October 1, 2011. With this shortened operating period, the team estimated that, the RVCH could be allowed to operate up to 624°F without exceeding the RIY limit. Further, 624°F was above any fuel assembly flow channel exit temperatures during this period and well above the licensee's best estimated head temperatures for operating cycle No.17. Because of these margins, the team concluded that the licensee had appropriately addressed the uncertainties in head temperature.

To understand the basis for the licensee's previous decision to apply the hot leg temperature as the "best estimate" for RVCH operating temperature, the team reviewed applicable licensee correspondence with the NRC, previous NRC inspection reports, and interviewed licensee staff. The team identified that both the licensee staff and NRC had previously questioned use of hot leg temperature to represent RVCH operating temperature. In 2004, the NRC reviewed licensee actions taken to comply with NRC Order EA 03-009 and questioned the licensee's basis for using the hot leg temperature to represent a "best estimate" vessel head temperature. The licensee described their basis for selecting hot leg temperature to represent a "best estimate" vessel head temperature for calculating EDY under NRC Order EA 03-009 as documented in the licensee's May 25, 2004 and August 18, 2004 letters (Serial Nos. 3045 (ADAMS Accession Number ML0414803520) and 3085 (ADAMS Accession Number ML042360716)). Additionally, in CR 04-03517, the licensee staff evaluated the possibility of using the head vent line temperature instrument to represent a more accurate head temperature. However, the licensee staff ultimately decided not to pursue a more accurate method of estimating head temperature following discussions with the EPRI project manager responsible for collecting and assessing PWSCC related data from US PWRs. Specifically, the licensee documented "it cannot be assured to properly correlate with any B&W-plant information if it is applied using temperature measured at a different location." This statement was in reference to the EPRI developed crack growth-rate correlations for B&W plants (reference MRP-75 "PWR Reactor Pressure Vessel Upper Head Penetrations Inspection Plan") that used hot leg temperature and not actual head operating temperature. Based on this information, the licensee decided that use of a more accurate head temperature derived from the head vent line could not be used to correlate with industry flaw growth models and thus would serve no purpose. The team considered this a reasonable basis with information known at this time and noted that even with a higher head operating temperature (e.g., 615 ^{o}F), no additional head examinations beyond those completed by the licensee, would have been required by NRC regulations. Therefore, the team concluded that no performance deficiency existed associated with the licensee's past decision to use hot leg temperature as a "best estimated" head temperature.

CRDM Nozzle Material

The RCT identified that random (e.g., poor) distribution of carbides present in the nozzle material was a significant contributor to the relatively early onset of PWSCC at Davis-Besse. Each of the 69 Alloy 600 nozzles at Davis-Besse were manufactured by B&W Tubular Products manufactured from two material heats. Sixty-eight nozzles were fabricated from Material Heat No. M7929 and only nozzle No. 2 was fabricated from Material Heat No. M6623. The specific method of fabricating the nozzle tubes was not recorded, but the licensee believed it would include rotary piercing or extruding over a mandrel followed by a mill anneal. To evaluate the specific material properties of heat M7929, the licensee removed ring and boat samples from CRDM nozzles No. 4 and 10 and performed metallurgical testing. The metallurgical analysis of boat samples removed CRDM nozzles No. 4 and No. 10 (Heat M7929) were documented in Westinghouse document RTU-MCE-10-36, "Final Report-Summary of Davis-Besse Unit 1 CRDM Nozzle Boat Samples Destructive Examinations." This report identified a random (e.g., poor) distribution of carbides present in both boat samples (Attachment 4, Picture 11). The desired microstructure for optimum resistance to PWSCC would have carbides located at the grain boundaries. To achieve this condition the mill annealing heat treatment temperature for SB-167 material should normally be in the range of 1850^{o}F to 1950^{o}F to put carbon into solution so that the carbides precipitate at the grain boundaries during cooling. However, very little annealing data was available for Material Heats M7929 or M6623. The only known requirement for this Alloy 600 material was to maintain the final annealing temperature above 1600^{o}F for a minimum of 10 minutes. Annealing temperatures as low as 1625^{o}F may have been used and this lower temperature (or time at this temperature) was insufficient to re-dissolve carbides. Therefore, the licensee's vendor report attributed the cause of the poor carbide distribution to annealing cycles, which were either not high enough in temperature and/ or not long enough for the given carbon level.

The two vessel nozzle Material Heats Nos. M7929 and M6623 were both fabricated by B&W to meet the ASME Code construction requirements. Material Heat M7929 was fabricated to meet the material requirements in SB-167 of ASME Code Section III 1968 Edition; summer 1968 Addenda and this heat had a yield strength of 43,000 pounds per square inch. Material Heat M6623 was fabricated to meet the material requirements in SB-167 of the ASME Code Section III 1971 Edition, winter 1971 Addenda and had yield strength of 43,692 pounds per square inch. The team reviewed certified material test records for these two heats of nozzle materials and confirmed that these materials conformed to the ASME Code specifications for SB-167 material.

b.5 Assessment of Root Cause and Corrective Actions

The licensee RCT followed the business practice Procedure (NOBP-LP-2011), which utilized the TapRoot methodology to identify potential causal factors through interviews and document reviews. Using this methodology, the RCT identified that the root cause for this event was a less than adequate perception of the risk of PWSCC susceptibility with the replacement RVCH resulting in inadequate identification, development, and implementation of interim actions to mitigate degradation prior to replacement with a PWSCC resistant Alloy 690 head. The RCT identified this root cause in part, because the licensee staff could have voluntarily taken more conservative actions sooner to preclude the PWSCC induced nozzle leakage. For example, the licensee could have elected to schedule volumetric examinations or head replacement prior to the 2010 outage. The team concluded that the licensee had established an adequate basis for this root cause, but did not consider this cause to represent a performance deficiency. Specifically, the team concluded that licensee staff had adequately considered relevant industry operating experience and completed all NRC required RVCH examinations so a performance deficiency did not exist. Hence, the team determined that licensee actions were reasonable and ultimately successful. In particular, although cracking resulted in minor CRDM leakage, the licensee identified the PWSCC in the nozzle and J-groove welds at an early stage through NRC required examinations, such that plant safety was not challenged.

The team identified that the RCT had not independently evaluated the site staff's knowledge and understanding of PWSCC, nor had the RCT independently evaluated the technical adequacy of the site training programs applicable to PWSCC. The RCT believed that FENOC staff knowledge and training on PWSCC were sufficient based on prior licensee staff "interactions" stemming from a 2006 and a 2008 root cause investigation. Additionally, the RCT believed that Alloy 600/690 Materials Management Program Owner was adequately trained based on his recent completion of a Job Familiarization Guideline. However, the team believed that without investigating or better understanding the current level of staff knowledge related to PWSCC, a broader underlying cause could possibly exist. The team concluded that reliance on other root cause investigations with licensee staff "interactions" to rule out staff knowledge or training as a potential cause was a weakness in the licensee's root cause determination process. This weakness was not considered a licensee performance deficiency, because the examinations of the RVCH met NRC requirements, which implied an adequate level of site staff knowledge and understanding in PWSCC.

The licensee RCT also identified a contributing cause for a less than adequate understanding of risk that also led to missed opportunities by the organization to recognize, more aggressively interrogate, and then validate actual RVCH operating temperatures and to correlate those temperatures to potential premature PWSCC development. The team did not identify a licensee performance deficiency associated with this contributing cause, based on review of CR 04-03517, which documented the licensee's basis for selecting hot leg temperature as a best estimated head operating temperature as discussed (4OA3.3.b.4) in this report.

The licensee assigned two corrective actions (CA) to prevent recurrence of the root cause. Corrective Action 10-73323-CA13 required the site to track to completion the installation of a new head made of PWSCC resistant materials; and CA 10-73323-CA14 required the creation of repetitive activities that monitor and calculate RVCH temperature and RIY every three months throughout operating cycle until RVCH was replaced. The team concluded that these actions were key actions for preventing recurrence of the direct cause (PWSCC), but it was not clear how these actions corrected the root cause related to a less than adequate perception of risk of PWSCC. The team also noted that the CAs were focused on RVCH instead of assigning broader actions to evaluate the extent of Alloy 600 components that might be impacted by the site's staff less than adequate perception of risk of PWSCC for the RVCH. For example, other Alloy 600/182/82 components have required repairs after identification of PWSCC at Davis-Besse and the RCT did not assess or assign actions to determine if these issues had any common link with the root cause identified. The RCT determined that no additional corrective measures were required beyond those already identified, because the action to replace the head would make any other actions unnecessary. The team agreed that the CAs assigned were sufficient to ensure a low likelihood of recurrence of PWSCC induced nozzle leakage in the head. Therefore, the team concluded that the licensee's root cause and CAs were adequate to meet NRC regulations.

b.6 Head Operational Service Limits Based on RIY and Crack Growth Analysis

To evaluate the allowable head service period before reaching the NRC required RIY limit identified in CC N-729-1, the licensee performed Calculation C-ME-099.99-013, "Effective Degradation Years (EDY) Calculation for Alloy 600/82/182 PWSCC Susceptibility Determination Related Applications." In this calculation, the licensee selected a "best estimated" RVCH temperature from the maximum fuel assembly exit temperatures that ranged from 612.4 ^{o}F thru 613.6 ^{o}F over six periods within Operating Cycle No. 17. Using these revised "best estimated" RVCH operating temperatures, the licensee determined that they would remain below 2.25 RIY limit from CC N-729-1 for the 593 effective full power days. Additionally, due to the RVCH temperatures being even higher, near 618°F in past operating cycles, the licensee determined that the RVCH would reach 12.1 EDY at the conclusion of Operating Cycle No. 17. The licensee considered calculation C-ME-099.99-013 results conservative because the RVCH "best estimate" input temperatures assumed were higher than predicted by the supporting calculation (AREVA document 51-9137401–000) and bounding inputs for cycle operational data were used (e.g., startup date of June 26, 2010, and 100 percent capacity factor over the entire cycle length). However, the licensee elected to limit operation to approximately 460 days and then replace the head based on the results of a crack growth evaluation and following interactions with NRC staff as discussed below.

To evaluate the head operating service life, before PWSCC induced leakage could recur, the licensee contracted a vendor to perform Calculation No. 1000422.401 "Crack Growth Evaluations of Control Rod Drive Mechanism Nozzle Penetrations at Davis-Besse Nuclear Power Station." This analysis predicted minimum crack growth times from the assumed initial flaw sizes to leakage, ranging from 2.3 to 21.1 years for nozzle cracks and from 1.6 to 10 years for weld cracks, using a variety of crack growth-rates and stress intensity factors. The minimum J-groove weld crack leakage time of 1.6 years was calculated for the 0-degree nozzle No. 1 location that had been repaired and thus was not applicable. At all other nozzle J-groove weld locations the minimum time to leakage increased and ranged up to 10 years. This analysis accounted for the increased reactor vessel head operating temperature (614.4 ^{o}F) and the effects of lack of fusion (LOF) flaws, which the licensee had identified in the J-groove welds (4OA3.4). In this analysis, the LOF flaws located in the mid-weld or lower weld area were determined to have an insignificant effect on minimum time to leakage.

Specifically, a LOF flaw located mid-weld or near the bottom (toes) of the J-groove welds could appear to reduce the through-wall leakage times with a shorter weld metal path to traverse. However, if a growing PWSCC crack were to intersect a LOF at those locations, the crack front would jump to the other side of the LOF flaw, but at this point it would require suitable conditions to initiate a new PWSCC flaw rather than continue with PWSCC growth. Due to the fast growth-rates through the material actively growing by PWSCC, the period of secondary initiation would be greater than the time for continued PWSCC growth through the potentially LOF flawed material. The licensee's vendor identified laboratory data that indicated a significant amount of time (1500 hours or greater) was required to begin actively growing PWSCC under a constant stress intensity factor. Therefore, the licensee's vendor identified that only the LOF flaws located at or near the triple point (root of the weld) need be considered in evaluating the limiting cases for through-weld PWSCC induced leakage. For welds with LOF indications, the limiting location was the nozzle No. 2 J-groove weld and the minimum crack growth time until leakage was 1.74 years. This calculation also evaluated the time required to generate wastage of the RVCH after CRDM nozzle leakage started. Specifically, the most conservative case run predicted an additional 1.2 years would be required for a nozzle crack to grow 1.0 inch beyond the top of the weld. The wastage calculations, based on this result, show that at least 2 years of additional time, from the inception of leakage, would be needed for structurally significant wastage of the RVCH to occur.

In calculation No. 1000422.401, the licensee's vendor applied a 95 percentile crack growth-rate for the nozzle base materials and a 75 percentile crack growth-rate for the J-groove welds as the maximum growth-rates expected based on the observed cracking and industry sources (references Materials Reliability Program: Crack Growth-Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Materials (MRP-55) Revision 1 and Materials Reliability Program: Crack Growth-Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82/182, and 132 Welds (MRP-115)). The licensee believed that these were conservative and appropriate PWSCC growth-rates. However, the prediction of PWSCC initiation time and propagation in CRDM nozzles and J-groove welds is difficult due to the uncertainty of numerous contributory variables including temperature, heat treatment, cold work, and residual stress. Additionally, repairs made during J-groove weld fabrication can have a significant effect on the as-welded residual stress, which in turn can have a substantive effect on PWSCC propagation.

Because of the uncertainties in the variables affecting PWSCC initiation and growth-rates, the NRC performed an independent conservative/bounding analysis of the minimum time to leakage with higher PWSCC growth-rates and a wider range of postulated PWSCC flaws in the J-groove weld and nozzles. The NRC analysis indicated that it was possible to have PWSCC induced leakage within a shorter time period than predicted by the licensee's calculation. Therefore, the team requested that the licensee quantify the uncertainty in the head operating temperature and other variables that affect calculation 1000422.401 or perform a more bounding analysis. The licensee elected to limit the current plant operating cycle to October 1, 2011, and then shutdown to replace the head. In Confirmatory Action Letter (CAL) 3-10-001, the NRC accepted the licensee's commitment to operate until October 1, 2011, and then shutdown to replace the head with a new head fabricated from PWSCC resistant materials. With this revised outage date, the RVCH would accumulate approximately 1.25 effective full power years (EFPYs) which represented a 28 percent reduction in service time before reaching the minimum time to leakage based upon the licensee's limiting PWSCC growth calculation. With this revised outage date, the team estimated that the licensee would have a maximum of 460 effective full power days of operation which represented a 22 percent reduction in cycle length from the allowed cycle length based on the RIY limit using the licensee's calculated RVCH operating temperature. Therefore, the team concluded that sufficient operating margins existed, to provide reasonable assurance that the likelihood for PWSCC induced leakage at the unrepaired CRDM nozzle and J-groove welds would remain low during the remaining planned RVCH service period. Additionally, to confirm crack growth-rates, the NRC intends to perform independent crack growth- rate tests on the nozzle materials removed from CRDM nozzle Nos. 4 or 10 (reference CAL 3-10-001-Adams Accession No. ML101740519).

.4 Review Current Examination Results and Monitor In-Progress Examination and Analysis Activities to Ensure they are adequately Conducted. Confirm Based on Review of the Examination Results, that the Licensee Has Identified Appropriate Nozzles for Repair and the Acceptability of Remaining Nozzles for Service.

a. Inspection Scope

The team observed and reviewed NDE activities performed on the RVCH and CRDM nozzles and J-groove welds to confirm that these examinations met NRC requirements and that appropriate acceptance criteria were applied for continued service. Specifically, the team observed the licensee and vendor staff performing the following:

- A BMV examination of the CRDM nozzle and vessel head surfaces;
- PT and ET examinations of the CRDM nozzle J-groove welds; and
- UT examination of the CRDM nozzles.

For ten CRDM nozzle locations, the team also performed an independent review of UT data to confirm the extent of examinations, application of flaw criteria and leakage path criteria and to confirm appropriate calibration checks were completed. The NRC also contracted with Pacific Northwest National Laboratory (PNNL) to perform an independent analysis of the Davis-Besse UT examination data. Specifically, PNNL performed a review of UT data collected for ten CRDM nozzle penetration locations and documented the results in a report that the team used to assess the licensee's capability to detect and size nozzle cracks.

The team also reviewed the overall examination results to evaluate the licensee's basis for identification of leaking nozzles and to assess the safety significance for past operating periods with PWSCCs in the nozzles and J-groove welds.

The NRC Inspection Procedure (IP) 71111.08 "Inservice Inspection" was partially completed for the Davis-Besse Nuclear Power Station (reference NRC IR 05000346/2010002). All Sections of IP 71111.08 were previously completed, except for Section 02.02 "Pressurized Water Reactor Vessel Upper Head Penetration Inspection Activities." The team's review of the RVCH examinations as discussed above fulfills the inspection requirements of Section 02.02 of IP 71111.08 and IP 71111.08 is complete.

b. Findings and Observations

In accordance with 10 CFR 50.55a(g)(6)(ii)(D) and CC N-729-1 the licensee was required to perform examinations of the RVCH and CRDM nozzles. The team confirmed through direct observation and review of records that the UT, BMV and PT examinations of the RVCH, CRDM nozzles and J-groove welds completed during the 2010 outage met these NRC requirements. Specifically, the extent of these examinations met CC N-729-1 requirements and the licensee applied examination equipment, procedures, and personnel qualified in accordance with the ASME Code as implemented by 10 CFR 50.55a.

Additionally, the team confirmed that the licensee had applied the applicable acceptance criteria to indications identified during these examinations. Specifically, the team performed a review of NDE records to confirm that CRDM nozzles and J-groove welds with crack indications were not returned to service.

The licensee determined that pressure boundary leakage occurred at CRDM nozzles Nos. 4 and 67 based on UT examination results. The team concluded that the licensee may have been over-reliant on the UT leakage path backwall pattern method used to identify nozzles with through-wall leakage, which could lead to an underestimate in the number of leaking nozzles. This conclusion did not represent a safety concern, because all nozzles with crack indications that could cause leakage were repaired prior to returning them to service.

Although cracking resulted in minor CRDM nozzle leakage, the team concluded that the licensee-identified PWSCC in the nozzle and J-groove welds at an early stage, such that plant safety was not challenged.

b.1 Examination Results

As a result of UT examinations, the licensee-identified 12 CRDM nozzles with indications "indicative of PWSCC." Eleven nozzles contained axially oriented PWSCC indications and CRDM nozzle No. 51 contained a circumferentially oriented PWSCC indication. For CRDM nozzle No. 4, the licensee identified three axial PWSCC indications (Attachment 4, Picture 7) and one of these indications also had a small circumferential component. The UT examinations of nozzles Nos. 4 and 67 also identified a backwall pattern indicative of through-wall leakage. The CRDM nozzles containing PWSCC indications including orientations and dimensions are recorded in (Table 3 of Attachment 3). Additionally, the UT examinations identified weld fusion zone indications (FZIs) in most of the CRDM nozzle locations and these were recorded on the UT data sheets if they extended 10 percent or more into the nozzle wall. The licensee performed additional characterization and evaluation of weld FZIs as discussed below. The licensee concluded that the weld FZIs were typical for this weld configuration and not structurally significant or connected to the wetted surface.

As a result of BMV examinations, the licensee initially identified deposits on several nozzles as documented in (Attachment 3, Table 1). The licensee then applied 60 pounds per square inch gage (psig) air source to blow away loose deposits/debris from the nozzle interface areas. Following application of this air cleaning, the licensee identified 14 CRDM nozzle locations with adherent boric acid deposits potentially indicative of leakage (Attachment 4, Pictures 1 through 3).

Of these, only nozzle No. 4 was identified with “popcorn” shaped deposits characteristic of “active leakage” and the remaining 13 CRDM nozzle locations were classified as “potential active leakage” because of the less definitive nature of the deposits.

As a result of PT and ET examinations on the wetted surface of CRDM nozzle J-groove welds, the licensee identified 12 J-groove weld locations with rejectable indications as identified in (Attachment 3, Table 2). The licensee performed PT examinations on the wetted surface of 11 CRDM nozzle J-groove welds, and the licensee identified four with rejectable indications and four with recordable indications. For three of the four J-groove welds with recordable PT indications, the licensee also performed ET examinations to confirm rejectable indications and added these nozzle locations to the repair list. For the nozzle No. 53 J-groove weld, the licensee removed the recordable PT indications with surface grinding. A followup PT examination of the nozzle No. 53 J-groove weld confirmed that the previous PT indications had been removed, so the licensee accepted this J-groove weld for continued service. The licensee also completed ET examination of the wetted surface of the J-groove welds for 51 CRDM nozzles that were not already designated for repair. This population excluded those J-groove welds subjected to a PT examination with satisfactory results. Based on ET examination, the licensee identified six J-groove welds with rejectable indications and two other J-groove welds (nozzles Nos. 1 and 3) with ET indications that met acceptance criteria, but elected to repair them. Specifically, J-groove weld at nozzle No. 1 had a small 0.16 inch diameter rounded indication and J-groove weld at nozzle No. 3 had a non-quantifiable indication at the weld to cladding interface that was confirmed by a visual examination.

Following CRDM nozzle repairs with the RCS at normal operating pressure and temperature, a licensee level III qualified VT-2 inspector completed a visual examination of the bare metal head surface through an access port in the service structure. No evidence of leakage was identified during this examination.

b.2 Evaluation of Examination Results

Cracking Safety Significance

The safety significance of a PWSCC depends on the crack location, size and if through-wall, time in service. PWSCCs that develop in the nozzle materials or J-groove welds are very tight (e.g. have very little width) such that a tortuous leakage path exists, typically resulting in very low leakage rates. However, if leakage is allowed to continue uncorrected for two or more years, substantive corrosion and loss of RVCH material can occur and begin to challenge the structural integrity of the RVCH (reference calculation 1000422.401 “Crack Growth Evaluations of Control Rod Drive Mechanism Nozzle Penetrations at Davis-Besse Nuclear Power Station.”) Additionally, circumferentially oriented PWSCC in the CRDM nozzle located at or above the top of the J-groove can also challenge the structural integrity of the nozzle and head. Industry reports - Material Reliability Program (MRP) 103 and MRP 110 (ADAMS Accession Nos. ML041680477 and ML041680506) and NRC Memorandum “Results of Independent Evaluation of Recent Reactor Vessel Head Penetration Cracking” (ADAMS Accession No. ML022400323) have determined that circumferential cracks in excess of 270 degrees can result in loss of structural integrity (e.g., the nozzle could potentially be ejected by reactor coolant pressure and initiate a loss-of-coolant accident).

The licensee-identified axial PWSCCs in several nozzles up to approximately 2 inches in length and for nozzle Nos. 4 and 67 these cracks breached the nozzle wall and/or J-groove weld resulting in minor leakage. Because these cracks were detected at an early stage, no significant corrosion or wastage occurred on the RVCH. The licensee also identified a small (e.g., < 15 degree extent) part-wall circumferentially oriented PWSCC in nozzle No. 51 located below the J-groove weld. At this location, even a full 360 degree crack would not result in nozzle ejection. Therefore, the team concluded that the licensee-identified PWSCC in the nozzle and J-groove welds at an early stage, such that plant safety was not challenged.

Leaking CRDM Nozzle Locations

The licensee determined that pressure boundary leakage occurred at CRDM nozzles Nos. 4 and 67 based on a UT backwall pattern indicative of leakage detected at these nozzles. During the BMV examination, the licensee had identified boric acid deposits indicative of leakage at nozzle No. 4, but not at nozzle No. 67. Nevertheless, the licensee concluded that nozzle No. 67 was leaking based on the presence of a UT leakage path pattern. The licensee stated that no boric acid was identified at nozzle No. 67 during the BMV examination because leakage was stopped by the tight interference fit between the nozzle and the RVCH. Based on review of UT data, the team noted that for nozzle No. 33, the vertical extent of the axial PWSCC traversed the height of the J-groove weld. A crack of this size could potentially provide a path for through-wall leakage and boric acid deposits potentially indicative of leakage were identified at nozzle No. 33 during the BMV examination (Picture 3 of Attachment 4). Nevertheless, the licensee concluded that nozzle No. 33 was not leaking because no UT backwall leakage pattern was detected at this location. Further, the licensee concluded that other than nozzle No. 4, the presence of boric acid deposits at other nozzle locations were inconclusive with respect to confirmation of active leakage. The licensee attributed the presence of boric acid at other nozzle locations to the transference of deposits generated at leaking nozzle No. 4. The licensee did not further investigate or document how deposits generated at nozzle No. 4 could migrate to other nozzle locations. The team noted that NRC has not approved an industry performance standard to evaluate the reliability of the UT leakage path method for detection of through-wall leakage. Therefore, the licensee may have been over-reliant on the UT leakage path backwall pattern method to confirm through-wall leakage. If the UT leakage path method was not reliable, it could have caused the licensee to underestimate the number of leaking nozzles. This did not represent a safety concern, because all nozzles with crack indications that could cause leakage (confirmed by UT, ET, and PT examinations) were repaired prior to returning them to service.

Weld Fusion Zone Indications (FZI)

During the UT examinations, the licensee-identified weld FZIs in most of the CRDM nozzle locations that were the result of slag inclusions, weld repairs, or voids caused by LOF. The UT examination cannot readily distinguish which of these three types of defects is causing the FZI. Of the three types of FZI defects, the team concluded that the LOF voids had the greatest potential to affect the structural or leakage integrity of the CRDM nozzles and J-groove welds. The LOF voids were confirmed to be present in the weld metal as documented in the licensee metallurgical analysis of the boat samples removed from CRDM nozzle No. 4 and 10 (Reference Westinghouse Document RTU-MCE-10-36, "Final Report-Summary of Davis-Besse Unit 1 CRDM Nozzle Boat Samples

Destructive Examinations"). Additionally, during the CRDM nozzle repairs, the team observed voids in the remnant J-groove welds that the licensee attributed to LOF.

The licensee used the UT data to plot the extent of the weld FZIs and documented the results in AREVA document 51-9137763-000 "Davis-Besse RFO 16 "J" Groove Weld Fusion Zone Indication Survey." This document recorded that 17 of the 45 unrepaired nozzles were free of weld FZIs. For the remaining 28 unrepaired CRDM nozzles with FZIs the licensee documented the number, area, and locations of each FZI. These weld FZIs were also present in the preservice records for each of these nozzles. The UT data analysis technique used to map the extent of the weld FZIs was validated on mockup test blocks. Specifically, the licensee's vendor documented the use of two EPRI mockup blocks (690 and H) that contained flat bottom holes to validate the UT analysis technique employed to map the weld FZIs. The team concluded that use of these mockups established a credible basis for the UT technique used to determine the extent of the weld FZIs.

In AREVA Document 51-9137763-000, the extent of J-groove weld-to-tube surface areas affected by FZIs was determined for each of the 28 unrepaired CRDM nozzles. The percentage of total weld surface area affected by FZIs ranged to a maximum of 5.1 percent for CRDM nozzle No. 44. The licensee determined that these weld FZIs did not affect the integrity of the vessel head because they were expected with the J-groove weld fabrication processes and the vessel head met all original construction Code acceptance criteria. Additionally, for the cracks or LOF voids that remained in the remnant J-groove welds of the repaired nozzles, the licensee completed a flaw growth analysis to demonstrate that these flaws did not affect leakage or structural integrity of the RVCH (reference FENOC letter dated April 21, 2010, ADAMS Accession No. ML101400402 and AREVA Calculation 32-9136508-002, ADAMS Accession No ML 101400406). However, for the unrepaired CRDM nozzles, the team questioned if the FZIs which included LOF voids, could shorten the time required for a PWSCC to traverse through the J-groove weld region and cause pressure boundary leakage. The licensee stated "If a PWSCC flaw were to link with a lack of fusion void, the time for the flaw to reach the annulus of the penetration could be shortened since the flaw does not have to sustain crack growth through the axial extent of the LOF void." The licensee identified that a LOF void located at the "triple point" would be the most limiting in terms of shortening the time for a PWSCC flaw to cause leakage. The team concluded that the licensee had established an adequate basis for evaluating LOF voids. Specifically, the licensee completed an analysis (Reference Vendor Structural Integrity Calculation No. 1000422.401) for the limiting LOF void, which occurred in the J-groove weld of nozzle No 2. This analysis was reviewed by the team in report Section 4OA3.3.

b.3 Examinations Required by NRC Regulations

Volumetric Examinations of CRDM Nozzles

To identify CRDM nozzles with flaw indications, the licensee performed UT with a time-of-flight diffraction technique. Each of the 69 CRDM nozzles was examined twice with different UT search units. The first examination utilized a UT blade probe sensitive to axial oriented flaws and the second examination utilized a UT blade probe sensitive to circumferentially oriented flaws. The procedure, equipment, and personnel used in these examinations were demonstrated through industry's Performance Demonstration Initiative (PDI), which is managed by EPRI. The PDI Program complied with the NRC mandated ASME Code Section XI Appendix VIII requirements for performance demonstration of UT equipment and personnel. The team confirmed that the equipment and personnel used for these inspections had certification records issued by the PDI Program applicable to the examinations completed on the CRDM nozzles.

The team observed licensee contractors acquiring and analyzing UT data for 10 CRDM nozzles. Based on this review, the team concluded that the licensee:

- Implemented adequate controls for locating penetrations;
- Achieved coverage for the head surface that met CC N-729-1 requirements;
- Used trained/qualified personnel; and
- Recorded flaw indications in accordance with the procedure.

For ten or more CRDM nozzle locations, the team reviewed electronically recorded UT data to confirm the extent of examinations, application of flaw criteria and leakage path criteria and confirmation of calibration checks. For these 10 nozzle locations, the NRC also obtained an independent review of UT data by the PNNL.

In "Evaluation of Ultrasonic Time-of-Flight Diffraction Data for Selected Control Rod Drive Nozzles from Davis-Besse Nuclear Power Station," the PNNL staff documented (Reference PNNL Report No. 19362) their independent review of UT data collected by the licensee's vendor at CRDM nozzle locations Nos. 4, 12, 19, 40, 42, 48, 57, 60, 66, and 67. The PNNL identified 3 indications consistent with cracking in nozzle No. 4 and sized these indications. The team compared the PWSCC indications in nozzle No. 4 sized by PNNL, to those sized by the licensee. The licensee's crack depth and length sizes measured were slightly larger than those recorded by PNNL, which indicated that the licensee had conservatively sized these cracks. For the remaining 9 nozzles, PNNL identified UT indications, but none were considered indicative of cracks that would result in leakage. The licensee-identified a PWSCC indication in nozzle No. 67 which was not considered a crack type flaw based on the PNNL staff review. The team discussed this result with the licensee's vendor UT analyst, who indicated that the potential leakage path data collected for that nozzle was the key factor in concluding that this nozzle contained a flaw/crack. Based on these results, the team concluded that the licensee had appropriately identified nozzles with crack indications.

<u>BMV Examination of the RVCH and CRDM Nozzles</u>

To identify boric acid deposits indicative of through wall-leakage, the licensee performed a BMV examination of the top surface of the head for the extent required by CC N-729-1 (Table 1 and Figure 1) using a remote camera mounted to a robotic crawler. This camera system and lighting was demonstrated at the maximum examination distance by visual resolution of 0.105 inch lower case alpha numeric letters to meet the requirements of CC N-729-1. The licensee performed verification of this visual resolution capability once per shift during the BMV examination and the team observed this visual resolution demonstration. Additionally, the team viewed the bare metal surface of the head through each of the ten openings in the service structure in order to confirm that the remote visual system provide a superior viewing capability over a direct visual examination conducted from the service structure openings.

The team reviewed video images of the BMV head examination records and observed the licensee acquiring video images using the remote camera system. Based on this review the team concluded that the licensee:

- Implemented adequate controls for locating penetrations;
- Achieved coverage for the head surface that met CC N-729-1, Figure 1 requirements;
- Used trained/qualified personnel; and
- Accurately recorded the location and nature of boric acid deposits.

<u>Surface Examinations of CRDM Nozzle J-Groove Welds</u>

The licensee performed surface examinations (PT or ET) of all J-groove welds on nozzles that were not subject to the half-nozzle repair. For 11 J-groove weld locations, the licensee applied a color contrast water-washable type PT process to identify flaws. The licensee selected these nozzle locations based on the results of the BMV and UT examinations. During the BMV examination the licensee-identified 14 CRDM nozzles with boric acid deposits potentially indicative of leakage. To determine if these deposits were the result of leakage caused by cracks originating in the J-groove welds, the licensee performed PT examinations. Specifically, for 9 CRDM nozzle J-groove welds with boric acid deposits indicative of potential leakage, the licensee completed a PT examination of the J-groove weld surface. For the remaining 5 nozzles with boric acid deposits indicative of potential leakage, the licensee did not perform a PT examination of the J-groove weld because these nozzles contained rejectable UT indications (e.g., cracks) in the nozzle base materials and were selected for repair. Additionally, the licensee elected to perform PT examination of two J-groove weld locations (No. 42 and No. 66) to confirm the UT examination results (e.g., no flaws).

The team observed the PT examination at seven of these J-groove welds locations. Based on this review, the team concluded that the licensee:

- Implemented adequate controls for locating penetrations;
- Established adequate lighting for visual resolution of indications;
- Performed an extent of examination that included full wetted surface of J-groove weld;
- Achieved dwell time for the penetrant and developer that met the procedure ranges and the ASME Code, Section V;
- Verified temperature of J-groove welds was within procedure and Code limits;
- Used trained/qualified personnel and Code qualified procedures; and
- Recorded relevant crack indications.

<u>Licensee-identified Error in Acceptance of the Nozzle No. 53 J-groove Weld PT Exam</u>

Following grinding to remove two minor surface indications, the licensee accepted the results of the final PT examination for CRDM nozzle No. 53 J-groove weld. Subsequently, while reviewing a picture of this final PT examination, the licensee identified a small rounded indication less than 1/32 inch in diameter that had not been recorded in the final PT examination report. The licensee performed additional grinding to remove this indication and completed another PT examination to accept the nozzle No. 53 J-groove weld. Additionally, the licensee re-reviewed pictures for each of the other acceptable J-groove weld PTs and did not identify any other deficiencies. The team reviewed pictures of the nozzle No. 53 J-groove weld and pictures of the final PT examinations complete on nozzles Nos. 11 and 12 J-groove welds to confirm that the examination records and results were acceptable. The failure of the licensee to initially identify the minor surface indications present in the PT examination of the nozzle No. 53 J-groove weld was a violation of 10 CFR Part 50, Appendix B, Criterion V "Instructions, Procedures and Drawings" of minor significance because the deficiency was corrected prior to returning this nozzle to service. The licensee also entered this deficiency into the corrective action system (CR 10-78081).

b.4 Licensee Initiative Examinations - Eddy Current Examinations (ET) of J-Groove Welds

The licensee elected to perform ET of the wetted surface of the J-groove welds for CRDM nozzles that were not already designated for repair. The team observed the licensee performing ET of these nozzles. Based on this review the team concluded that the licensee:

- Implemented adequate controls for locating penetrations;
- Established adequate surface contact to avoid liftoff signals on the ET probe;
- Performed an extent of examination that included full wetted surface of J-groove weld;
- Used trained/qualified personnel;
- Used a procedure qualified as a low rigor examination in accordance with the ASME Code Section V, Article 14; and
- Recorded relevant crack indications.

b.5 Post Repair Visual Examination of RVCH Surface at Normal Operating Pressure

On May 24, 2010, the NRC staff held a teleconference with licensee staff to clarify information contained in the FENOC letters dated April 1, 2010, and May 17, 2010, where the licensee submitted the half-nozzle repair relief request (RR) for the CRDM nozzles. In response to an NRC request for additional information on these submittals, the licensee stated that CC N-416-3 would be used to satisfy pressure testing requirements subsequent to the CRDM nozzle repair. The licensee indicated that the VT-2 examination would be conducted from outside the service structure of the reactor pressure vessel with insulation installed such that no direct view of the bare metal surface of the head in the areas that were repaired would be visible. In general, the NRC has accepted this type of indirect VT-2 examination. However, the NRC staff questioned if this method was appropriate for the CRDM nozzle repairs at Davis-Besse given the following:

- A relatively large number of repaired CRDM nozzles may increase the possibility for fabrication defects that are undetected by NDE;

- The initial unsuccessful repair attempts on CRDM nozzle No. 4 may have increased the possibility for fabrication defects; and
- The inability of a VT-2 examination conducted from the outside service structure to detect small leakage from a through-wall fabrication defect in a repaired nozzle.

For these reasons, the NRC requested the licensee conduct a visual examination of the bare metal head surface through one or more service structure access ports during the system leakage test at normal operating pressure to provide an increased level of confidence in the nozzle repairs. Alternatively, the NRC requested the licensee justify why the proposed indirect VT-2 examination was acceptable.

On May 28, 2010, the licensee provided an electronic mail response clarifying this issue (reference ADAMS Accession Number ML101520088). In this response, the licensee stated "The CRDM penetrations are located within the service structure below the reactor vessel head's insulation package. Access to this area is available through inspection ports, however, during the system leakage test, the RCS is at normal operating temperature and pressure (approximately 532 ^{o}F and 2150 psig) with all insulation in place as permitted by IWA-5000. This insulation covers the inspection ports. Opening the inspection ports would require removal of the insulation and the hot inspection ports. Air emitted from the ports could approach 500 ^{o}F. Inspection would be difficult and access would require personnel to be in close proximity to the hot surfaces creating an industrial safety hazard. The benefit of this direct visual examination is considered to be limited given the additional level of NDE and pressure test visual examination noted above. However, in addition to the VT-2 examination discussed above, FENOC intends to perform a visual inspection for leakage from the reactor vessel head through an inspection port to the extent that access and environmental conditions permit."

On June 26, 2010, with the RCS at normal operating pressure and temperature a licensee level III qualified VT-2 inspector completed a visual examination of the bare metal head surface through an access port in the service structure. The licensee selected the service structure access port that provided the best view of the largest number of repaired nozzles, which included CRDM nozzle No. 4. The team observed this examination and confirmed that no evidence of leakage was detected.

.5 <u>Evaluate the Adequacy of the Repair Activities and Monitor Implementation. Confirm That the Repair Implemented Complies with NRC Requirements.</u>

a. <u>Inspection Scope</u>

On April 1, 2010, the licensee submitted RR A34 to the NRC to support repairs to the CRDM nozzles and J-groove welds (FENOC Letter L-10-099 – ADAMS Accession No. ML100960276). In RR A34, the licensee requested NRC approval to implement alternatives to ASME Code repair requirements. Specifically, the licensee proposed use of a "half-nozzle" repair technique (Attachment 4, Pictures 8 and 9) that provided for an inside diameter temper bead weld to restore the pressure boundary of these degraded nozzles. The half-nozzle repair process steps included: 1) roll expansion of the CRDM nozzle within the head; 2) removal (by machining) of the lower nozzle section; 3) PT examination of the machined head surface; 4) fabrication of a new weld (Alloy 52M) to attach the machined CRDM nozzle to the RVCH; 5) final machining of repair weld and CRDM nozzle bore; 6) PT and UT of the attachment weld; and 7) water jet remediation on the inside wetted surface of the expanded portions of the CRDM nozzle susceptible to PWSCC. This repair technique exposed a portion of the carbon steel nozzle bore which when returned to service would remain in contact with the reactor coolant. Although, the exposed carbon steel is subject to corrosion by the reactor coolant, the licensee determined that the material loss expected over the remaining service life of the head was inconsequential. Specifically, the licensee-identified that the general corrosion rate expected for the head material exposed to reactor coolant would be 0.0035 inches per year (reference FENOC Letter L-10-143 - ADAMS Accession No. ML101400402).

On April 16, 2010, and April 21, 2010, the licensee submitted supplemental information for their original relief request including J-groove weld flaw evaluations to support RR A34 (ADAMS Accession Nos. ML101110149 and ML101400402). On May 17, 2010, the licensee revised and updated the original relief request to address changes in repair methods applicable to CRDM nozzle No. 4 and to resolve staff questions (FENOC Letter L-10-143 - ADAMS Accession No. ML101400402). On May 28, 2010 (ADAMS Accession No. ML101520113), the licensee provided the results of the final NDE for the repaired CRDM nozzle No. 4 and the NRC provided verbal approval to RR A34 on June 4, 2010 (ADAMS Accession No. ML101600147). Additionally, the team and members of NRR staff observed the licensee's repair vendor applying the "half-nozzle" repair technique on a vendor mockup facility located in Lynchburg Virginia prior to applying this repair method at the Davis-Besse site.

The team reviewed the licensee's work instructions and work orders that controlled the CRDM nozzles repairs. The team performed on-site observation of the nozzle repair activities during each phase of the repair process including the post repair NDE used to accept the repaired nozzles for continued service. The team evaluated these activities to confirm that the nozzle repairs met the NRC approved repair method as described in RR A34. Specifically, the team observed and reviewed records for the implementation of the CRDM nozzle repairs to determine if:

- The licensee followed qualified weld procedures and used weld machine operators qualified in accordance with the ASME Code Section IX;
- The licensee installed weld filler materials traceable to Certified Material Test Reports;
- The licensee followed NDE procedures and used qualified examiners for repair weld examinations accordance with the ASME Code; and
- The licensee performed the correct type and extent of NDE and applied acceptance criteria in accordance with the approved RR A34.

b. Findings and Observations

The licensee-identified a total population of 24 CRDM nozzles with cracks in the J-groove weld or nozzle and repaired them with the "half-nozzle" repair technique approved by the NRC. The licensee completed the CRDM nozzle repairs from inside containment with the RVCH set on the normal head storage stand used during refueling outages. The equipment (e.g., machining tools, welding machines) used for these repairs was installed in each CRDM nozzle with a remote controlled robotic operator located under the RVCH. The repairs were typically controlled and monitored from a remote trailer location located just outside the site protected area.

The team observed the licensee performing each of the CRDM nozzle repair steps for three or more CRDM nozzle penetration locations. The repair welds were completed as planned except at CRDM nozzle No. 4. For this nozzle location, the licensee encountered contaminants, which affected the viscosity of the weld material puddle, such that the initial weld installation attempts failed. The licensee changed the repair process to incorporate Alloy 82 weld material and higher weld heat inputs to successfully complete installation of the weld. Based on direct observations and review of associated records, the team confirmed that the CRDM nozzles were repaired in accordance with the NRC approved relief request and thus suitable for return to service.

The team identified two findings related to the licensee deficiencies in implementation of a remote PT process used to confirm the acceptability of the CRDM nozzle repairs. Additionally, the team identified a finding associated with the licensee's use of an unqualified weld procedure for the repair to CRDM nozzle No. 4. The licensee implemented prompt corrective actions for these deficiencies to restore the affected CRDM nozzles to compliance with the NRC approved RR A34.

b.1 Roll Expansion and Machining Repair Steps

For the first repair step, the licensee used a remote controlled hydraulic expansion tool inserted from below the nozzle to expand the CRDM nozzle from the inside surface. The purpose of this step was to stabilize the nozzle to prevent movement when the nozzle is separated from the RVCH head J-groove weld during the follow-on machining operation. The team observed portions of the CRDM nozzle expansion process and no deviations from the work instructions were observed. The licensee's vendor completed this step of the operation without equipment problems or other repair challenges.

For the next repair step, the licensee installed a remote controlled hydraulic machining tool to the lower end of the CRDM nozzle protruding beneath the RVCH. The CRDM nozzle end base material was removed by the machining tool to a location near the mid-point of the bore within the RVCH. This process separated the nozzle from the original J-groove weld and removed a small amount of material from the RVCH by increasing the original head bore diameter. The team observed portions of this machining operation and reviewed the licensee's resolution to minor dimensional errors caused by the machine tool slippage as discussed below.

Machining Process Minor Dimensional Errors

Following machining, the licensee's contractor identified that the base material of CRDM nozzles Nos. 10, 43 and 51 had not been removed to the target depth (e.g., machined too shallow). Specifically, nozzle Nos. 10, 43, and 51 were out of tolerance by 0.008 inch, 0.042 inch, and 0.027 inch respectively. The licensee documented and evaluated this condition in CR 10-75030 and CR 10-75182 and accepted these nozzles as-is, based on the ability to maintain the minimum distance between the final repair weld to the original J-groove weld. Additionally, nozzle No. 4 was machined too shallow and was out of tolerance by 0.051 inch. The licensee corrected this error by re-machining the nozzle to achieve the correct target depth. The cause of these shallow cuts was attributed to slippage of the 212B machining tool anchor during the cutting process.

For nozzle No. 61 the licensee's contractor identified that the mechanical hard stop used to control the machining depth was set incorrectly. This error was detected by a non-procedure driven re-measurement to confirm the hard stop position. Absent this check, the machining operation would have removed an extra 0.800 inch of nozzle material, which would have required extensive reanalysis or change in the planned repair process. Fortunately, this did not occur and the calculations used to set the hard stop location were re-performed with licensee Quality Assurance staff providing an independent verification of the correct hard stop location. The licensee also performed independent reviews of all machining related dimensions completed by their contractor.

Following machining, the licensee's contractor identified that the weld preparation surface and/or the maximum bore diameter dimensions were out of specification for nozzles Nos. 1, 21, and 40. For nozzle No.1, the bore diameter exceeded the maximum tolerance by 0.049 inch and the weld preparation surface was machined below the minimum depth by 0.073 inch. For nozzle 40, the bore diameter exceeded the maximum tolerance by 0.063 inch and the weld preparation surface was machined below the minimum depth by 0.022 inch. For nozzle 21, the weld preparation surface was machined below the minimum depth by 0.049 inch. The cause of these machining errors was again attributed to slippage of the machining tool anchor during the cutting process. The licensee performed an evaluation in CR 10-76457 and accepted these nozzles as-is, based on the ability to maintain the minimum distance between the final repair weld to the original J-groove weld. Although the bore diameters for nozzles Nos. 1 and 40 were slightly outside of the machining tolerances established by the vendor, the as machined dimensions were bounded by the maximum bore diameter specified on the vendor's repair drawing (No. 02-9134305E-004).

The team was concerned with the frequency of repair process machining errors, which occurred under the quality assurance program implemented by RVCH repair contractor. In response, the licensee's quality control (QC) inspectors implemented increased field observations with focus on error likely conditions for the vendor repair steps. Specifically, the licensee's QC staff reviewed vendor established critical repair steps and incorporated these steps into the scope of QC field observations and monitoring activities. As of June 1, 2010, the licensee QC staff had performed over 125 field observations of the vendor repair activities. The team considered the licensee controls and corrective actions adequate to resolve these issues.

b.2 PT of the Machined Head Bore Surface Step

For the next repair step, the licensee completed a PT examination of the machined surface of the head bore and the machine beveled nozzle end weld preparation surface. To accomplish this step, the licensee's vendor inserted a PT tool mounted at the top of the CRDM nozzle flange and operated by staff located on the platform above the service structure. The team observed staff performing this remote PT process and identified that the licensee implemented an unqualified PT procedure as discussed below.

b.2.1 Unqualified PT Procedure For Nozzle Repairs

Introduction: The team identified a NCV of 10 CFR Part 50 Appendix B, Criterion IX for the licensee's failure to use a NDE procedure qualified in accordance with applicable Codes and Standards for detection of flaws in CRDM nozzle repairs. Specifically, the licensee failed to ensure that Procedure 54-ISI-244-10 "Liquid Penetrant Examination of

Reactor Vessel Head Penetrations from the Inside Surface," contained a maximum time limit for application of water-wash.

Description: On April 14, 2010, the team identified that the licensee failed to use a qualified PT procedure to detect flaws in the CRDM nozzle repairs.

The licensee's vendor applied procedure 54-ISI-244-10 to examine: the pre-weld machined surface of the vessel head penetration material, the machined surface of the CRDM nozzle base material, and the final machined surface of the CRDM repair weld. These examinations were intended to detect fabrication flaws so that they could be corrected prior to placing the repair weld in service. After machining steps and following application of penetrant on nozzles Nos. 10 and 61, the team identified that the duration of the 40 psig water-wash application varied from 4 to 12 minutes. The team was concerned that the licensee was not controlling the water-wash time, which could render the PT examinations ineffective. The licensee had approved the vendor procedure 54-ISI-244-10, but failed to ensure that this procedure incorporated controls for the maximum time that the 40 psig water-wash could be applied to the surface after penetrant application. Excessive time in application of the water-wash would remove penetrant material from flaws so that subsequent developer application would not draw out penetrant fluid and reveal fabrication flaws. Specifically, the PT procedure may not have been adequate to detect voids or cracks in the base material or repair welds.

The team determined that without establishing a maximum time for application of the water-wash, the licensee had not met the ASME Code Section V, Article 6, Paragraph T-644 requirement that stated "After the specified penetration time has elapsed, any penetrant remaining on the surface shall be removed, taking care to minimize removal of penetrant from discontinuities" Further, Article 6, Paragraph T-610 stated that SE -165 "Standard Practice for Liquid Penetrant Inspection Method," may be considered for PT procedures. Paragraph 6.5.2.4 of SE-165 stated "Avoid overwashing; excessive washing can cause penetrant to be washed out of discontinuities." In this example, the licensee did not limit the water-wash time to ensure that excessive washing did not remove penetrant from discontinuities (e.g., cracks/flaws).

The licensee subsequently changed Procedure 54-ISI-244-10 to incorporate a maximum time limit of 10 minutes for the water-wash application time. The 10-minute limit was based on a demonstration PT examination completed on a mockup test block where a 15-minute water-wash step was applied. This demonstration bound the maximum water-wash time applied at any nozzle and was successful in identifying crack indications on a quenched cracked aluminum block mounted to the inside of a plastic pipe. The team also confirmed that the aluminum block used for this demonstration was fabricated in accordance with requirements of the ASME Code, Section V, Article 6, Paragraph T-653.2 "Liquid Penetrant Comparator."

Analysis: The team determined that the licensee's failure to ensure that a PT procedure for acceptance of the CRDM nozzle repairs met the applicable Code requirements was a performance deficiency that impacted the Initiating Events Cornerstone attribute of Equipment Performance. The team evaluated this performance deficiency against the examples of minor and more than minor findings identified in IMC 0612, Appendix E, and determined that there were no sufficiently similar examples. Specifically, this was not considered a work in progress deficiency, because the licensee had completed several nozzle PT examinations using the unqualified PT procedure and no other quality assurance measure existed to detect this error prior to placing the affected nozzles in service.

The team determined that this finding was more than minor because if left uncorrected, the failure to use a qualified PT procedure could become a more significant safety concern. Absent NRC identification, the licensee would not have controlled the maximum times used to wash the penetrant materials off repair weld surfaces. Excessive wash time could have resulted in failure to detect fabrication flaws such as voids and cracks. Undetected cracks returned to service in the repair welds would place the RVCH at increased risk for through-wall leakage and/or nozzle failure. Therefore, this finding adversely affected the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions.

The team completed a significance determination, in accordance with Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 0609.04, "Phase I - Initial Screening and Characterization of findings," Table 4a for the Initiating Events Cornerstone. The issue was corrected promptly, no cracks were returned to service, and the team answered "no" to the Phase I screening question that asked assuming the worst case degradation would the finding result in exceeding the Technical Specification limit for any reactor coolant system leakage. Therefore, the finding screened as having very low safety significance (Green). This finding had a cross-cutting aspect in the area of Human Performance, Work Practices per IMC 0310 (Item H.4(c)) because the licensee did not provide adequate supervisory and management oversight of work activities including contractors such that nuclear safety was supported. Specifically, the licensee failed to provide an adequate oversight in the review and acceptance of the unqualified vendor PT Procedure 54-ISI-244-10. The team concluded that this was the primary cause of the finding based upon discussions with licensee and vendor staff.

<u>Enforcement</u>: Title 10 CFR Part 50 Appendix B, Criterion IX "Control of Special Processes" required in part, that measures shall be established to ensure that special processes, including nondestructive testing are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

The 1989 Edition of the ASME Code Section V, Article 6, Paragraph T-644, stated in part "After the specified penetration time has elapsed, any penetrant remaining on the surface shall be removed, taking care to minimize removal of penetrant from discontinuities."

Contrary to the above, as of April 14, 2010, the licensee had not established an NDE procedure qualified for detection of flaws in accordance with applicable Codes and Standards. Specifically, Procedure 54-ISI-244-10 "Liquid Penetrant Examination of Reactor Vessel Head Penetrations from the Inside Surface," did not control the maximum time for application of water-wash and thus did not "take care to minimize removal of penetrant from discontinuities." Failure to use a qualified NDE procedure appropriate to the circumstance is a violation of 10 CFR Part 50 Appendix B, Criterion IX. Because of the very low safety significance of this finding and because the issue was entered into the licensee's corrective action program (CR 10-75709), it is being treated as a NCV, consistent with Section VI.A.1 of the Enforcement Policy (November 28, 2008) (NCV 05000346/2010008-01).

b.3 Weld Installation Step

For the next repair step, the licensee used a remote controlled welding machine to install the repair weld at the lower beveled end of the nozzle inside the head bore. This new pressure boundary weld connected the nozzle to the RVCH and was fabricated from Inconel Alloy 52M material. The Alloy 52M weld material was selected because it was much less susceptible to PWSCC than the weld materials (Inconel Alloy 82 and 182) used in the original J-groove welds. The team observed licensee vendor staff fabricating the new repair welds which were completed as planned except at nozzle No. 4. For this nozzle location, the licensee encountered contaminants, which affected the viscosity of the weld material puddle, such that the initial weld installation attempts failed. The licensee changed the repair process to incorporate Alloy 82 weld material and higher weld head inputs to complete installation of the weld at nozzle No. 4. The licensee updated their original Code RR A34 to reflect these changes and on June 4, 2010, the NRC approved the updated version of RR A34.

b.3.1 Weld Installation Process Changes for CRDM Nozzle No. 4

At nozzle No. 4, during welding of layer No. 1, weld pass No. 2, the team observed bubbling/spattering in the weld puddle. This weld pass was located at the “triple point” which is the junction point where the low alloy steel head, the Alloy 600 nozzle and the first Alloy 52M weld bead intersect. Subsequently, during application of layer No. 2, the licensee’s contractor identified and documented in CR 10-75552 unfused weld material at the triple point which they believe was due to boric acid/corrosion product contamination coming from the annulus gap between nozzle and head bore. In an update to RR A34, the licensee stated that the contaminants affected the viscosity of the weld material puddle and that the contaminants were not specifically known, but thought to be boric acid and corrosion products as a result of RCS leakage through the CRDM nozzle No. 4 crevice. Further, the licensee stated that heat from the welding at the crevice or triple point location drew contaminants out of the crevice into the weld puddle. The contractor performed grinding to remove the original weld passes and attempted another unsuccessful weld pass. This weld pass did not properly fuse for approximately 220 degrees of the weld. The licensee suspended weld activities for nozzle No. 4 and changed repair plans.

The licensee’s first revised repair plan included machining the head bore and CRDM nozzle bevel to remove the weld metal from the failed repair attempt and performing a weld repair at the triple point with Alloy 82 weld filler material. The team confirmed that WP3/43/F43TBSCa3-005 “Machine Temper Bead GTAW [Gas Tungsten Arc Welding]” was qualified in accordance with the ASME Code Section IX for the use of Alloy 82 weld filler materials and for the weld repair base materials. Although, the Alloy 82 filler material provided better weldability, it was potentially susceptible to PWSCC, so the repair design included covering the Alloy 82 material with several layers of Alloy 52M weld filler metal to preclude PWSCC. However, during attempts to install the Alloy 82 weld layer pass at the triple point, it again failed to fuse properly. The team observed that the weld pass bead at the triple point split upon cooling. The licensee attributed this weld failure to the excessive gap which now existed between the nozzle bevel and head bore caused by the welding heat applied during the first unsuccessful weld attempts. The licensee subsequently removed this unsuccessful weld by machining the head bore and nozzle end bevel. After completing this operation, the licensee could not achieve a satisfactory PT of the nozzle result due to “bleedout” of penetrant from behind the nozzle. Specifically, the excessive bleed out from the triple point crevice resulted in inadequate interpretation on adjacent areas of the nozzle bevel and bore surfaces in a circumferential band approximately 3/8 inch wide on the nozzle bevel surface and 3/8 inch on the bore surface adjacent to the crevice. Subsequently, the licensee changed RR A34 for the repair process to substitute a visual VT-1 examination of the nozzle bore for nozzle No. 4, instead of a PT examination. The NRC reviewed the licensee's change to RR A34 and determined that the proposed alternative was acceptable because it provided reasonable assurance of structural integrity of the welds (reference ADAMS Accession No. ML101600147).

The licensee's next repair plan included use of a higher weld heat with a greater weld metal deposition rate to bridge the gap which existed near the triple point from previous repair attempts. The licensee's vendor had identified a procedure qualification record (PQR) with higher weld heat inputs that had previously been used to support similar CRDM nozzle repairs at another utility. However, this PQR was not applicable to Davis-Besse because it did not use the same base metals as would be used for the Davis-Besse repairs. The licensee stated that they intended to proceed "at risk" with weld repairs using the "unqualified" weld procedure and then complete a PQR to demonstrate and qualify the procedure to meet CC N-638-1 and the ASME Code Section IX after the weld repair. The team informed the licensee that this activity would be contrary to10 CFR Part 50, Appendix B, Criterion IX, which required that welding be performed in accordance with weld procedures qualified in accordance with applicable Codes. After consideration of the NRC position, the licensee suspended weld repairs on CRDM nozzle No. 4 until the weld procedure was qualified in accordance with the ASME Code Section IX requirements. The licensee subsequently completed PQR 7295-000 to qualify WP3/43/F43TBSCa3-005 for a higher heat input planned for weld repairs on nozzle No. 4. The licensee implemented PQR 7295-000 via PS0140-002; Procedure Supplement for use in Conjunction with CRDM Penetration 4 Procedure Supplement PS01390 for OI-0031, which modified Weld Procedure (WP)3/43/F43TBSCa3-005 for repair welding to allow higher weld heat inputs and Alloy 82 weld filler metal on nozzle No. 4 only. With the increased heat input allowed by PS0140-002 and use of Alloy 82 weld material, the licensee was successful in completing the weld repair to CRDM nozzle No. 4. The licensee covered the Alloy 82 with Alloy 52 in the final weld repair for CRDM nozzle No. 4 to ensure that the repair weld would remain resistant to PWSCC. However, the team subsequently identified that portions of this repair were not completed with a qualified weld procedure as discussed in the following report Section.

b.3.2 Unqualified Weld Repair Applied at CRDM Nozzle No. 4

Introduction: The team identified a NCV of 10 CFR Part 50 Appendix B, Criterion IX for the licensee's failure to perform repair welding on CRDM nozzle No. 4 using a qualified weld procedure. Specifically, the licensee failed to ensure that the weld procedure supplement PS0140-002 controlled heat input to less than that demonstrated in the supporting weld PQR.

Description: On June 8, 2010, the team identified that the licensee failed to use a qualified weld procedure for repairs completed to CRDM nozzle No. 4.

In accordance with the NRC approved RR-A34 for the CRDM nozzle repairs, the licensee had committed to follow CC N-638-1. This CC required that weld repair layers 3 and higher be deposited with a heat input that did not exceed the heat input used in the PQR. A PQR is a record of the welding data used to weld a test coupon and it contains a record of the mechanical tests (impact, tensile or bend tests) performed on this weld sample. The PQR is used to “qualify” a welding procedure such that if the procedure controls welding variables within ASME Code limits, the material properties of a weld produced by a “qualified” weld procedure will be similar to those demonstrated in the PQR sample (e.g., toughness, ductility, or strength). Further, heat input must be controlled with the limits demonstrated in the PQR to meet the weld procedure qualification requirements of the ASME Code Section IX, Articles QW-256 and QW 409.1 for the GTAW process used in this repair.

During weld repairs to nozzle No. 4, the licensee experienced several failed attempts to complete this weld. To resolve this issue, the licensee approved a vendor WPS supplement PS0140-002 “Procedure Supplement for use in Conjunction with CRDM Penetration 4 Procedure Supplement PS01390 for OI-0031” to supplement the WP3/43/F43TBSCa3-005 “Machine Temper Bead GTAW,” which allowed higher weld heat input limits. The team identified that PS0140-002 was not a qualified welding procedure because it allowed welding for weld repair layers three and higher with heat inputs that exceeded those used in the supporting PQR 7295-000. Further, the team identified that repair weld layers Nos. 3 through10 had been installed with heat inputs approximately 2.5 percent higher than that supported by PQR 7295-000. Because, the repair weld heat input exceeded that in PQR 7295-000, the repair weld was not “qualified” in accordance with the ASME Code Section IX. The team was concerned that this unqualified repair weld lacked demonstration tests to confirm that the material properties (e.g., toughness, ductility, or strength) were adequate for return to service.

To restore compliance with the ASME Code and demonstrate that an adequate repair weld had been fabricated, the licensee completed a new weld coupon, tested the coupon, and documented the results in a new PQR. The team reviewed the revised Procedure Supplement PS0140-003 “Procedure Supplement for use in Conjunction with CRDM Penetration 4 Procedure Supplement PS01390 for OI-0031” and new supporting PQR; PQ7296-00. The PQR; PQ7296-00 recorded heat inputs for the new weld coupon that bound the heat input used for the weld repairs completed on CRDM nozzle No. 4 and the weld coupon test results demonstrated the weld properties were acceptable.

Analysis: The team determined that the licensee’s failure to apply a qualified weld procedure for the nozzle No. 4 weld repair was a performance deficiency that impacted the Initiating Events Cornerstone attribute of Equipment Performance. The team evaluated this performance deficiency against the examples of minor and more than minor findings identified in IMC 0612, Appendix E and determined that there were no sufficiently similar examples. Specifically, this was not considered a work in progress deficiency, because the licensee had completed the weld with the unqualified weld procedure and no other quality assurance measure existed to detect this error prior to placing the affected nozzle in service.

The team determined that this finding was more than minor because if left uncorrected, the failure to use a qualified weld procedure could become a more significant safety concern. Absent NRC identification, the licensee would not have completed a Code qualified weld repair on nozzle No. 4 prior to returning the RCVH to service. The repair weld lacked qualification tests to demonstrate that the mechanical properties (toughness, ductility or strength) were adequate, which could have placed the RVCH at an increased risk for through-wall leakage and/or nozzle failure. Therefore, this finding adversely affected the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions.

The team completed a significance determination, in accordance with Inspection Manual Chapter 0609, “Significance Determination Process,” Attachment 0609.04, “Phase I - Initial Screening and Characterization of findings,” Table 4a for the Initiating Events Cornerstone. The issue was corrected promptly, the unqualified repair weld was not placed in service, and the team answered “no” to the Phase I screening question that asked assuming the worst case degradation would the finding result in exceeding the Technical Specification limit for any reactor coolant system leakage. Therefore, the finding screened as having very low safety significance (Green).

This finding had a cross-cutting aspect in the area of Human Performance, Work Practices per IMC 0310 (Item H.4(c)) because the licensee did not provide adequate supervisory and management oversight of work activities including contractors such that nuclear safety was supported. Specifically, the licensee failed to provide an adequate oversight in the review and acceptance of the unqualified vendor Weld Procedure Supplement PS0140-002. The team concluded that this was the primary cause of the finding based upon discussions with licensee and vendor staff.

<u>Enforcement</u>: Title 10 CFR Part 50 Appendix B, Criterion IX “Control of Special Processes” required in part, that measures shall be established to ensure that special processes, including welding are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

Paragraph 3(c) of the ASME CC N-638-1; dated February 13, 2003, stated in part “Subsequent layers shall be deposited with a heat input not exceeding that used for layers beyond the third layer in the procedure qualification.”

Contrary to the above, on May 23, 2010, the licensee deposited subsequent layers (No’s 3-10) on the CRDM nozzle No. 4 repair weld using a heat input (27,500 kilojoules per inch) that exceeded that used for layers beyond the third layer as recorded in PQR 7295-000 (e.g. 26,933 kilojoules per inch). Failure to use a qualified weld procedure in accordance with the applicable Code is a violation of 10 CFR Part 50 Appendix B, Criterion IX. Because of the very low safety significance of this finding and because the issue was entered into the licensee’s corrective action program (CR 10-77957), it is being treated as a NCV, consistent with Section VI.A.1 of the Enforcement Policy (November 28, 2008) (NCV 05000346/2010008-02).

b.4 <u>Repair Weld Machining and NDE Steps</u>

For the next repair step, the licensee used a remote controlled hydraulic machining tool to remove material from the inner nozzle bore and repair weld to provide a suitable surface for NDE. The team observed portions of the final nozzle machining process and no deviations from the work instructions were observed. The licensee’s vendor completed this step of the operation without equipment problems or other repair challenges.

For the next repair step, the licensee completed a PT examination of the machined surface of the head bore and the final repair weld inside surface. To accomplish this step, the licensee's vendor inserted a remote PT tool mounted at the top of the CRDM nozzle flange and operated by staff located on the platform above the service structure. The purpose of this PT examination was to confirm that the final weld was free of fabrication defects. The team observed these PT examinations and reviewed the final PT records to confirm that the licensee had appropriately applied the acceptance criteria. The team identified a finding associated with the licensee's failure to observe the complete examination area during the PT examination as discussed in the following report section.

For the next repair step, the licensee completed a UT examination of the repair weld from the inside nozzle bore surface. The UT equipment was inserted up inside the nozzle using a remote controlled robotic delivery tool positioned underneath the RVCH. The purpose of this UT examination was to confirm that the final weld was free of fabrication defects. The team observed portions of this UT examination and no deviations from the work instructions were observed. Additionally, the team reviewed the final UT records to confirm that the licensee had appropriately applied the acceptance criteria. The licensee's vendor completed this step of the operation without equipment problems.

b.4.1 Inadequate Procedure for Viewing of Remote PT on Nozzle No. 61 Repair Weld

Introduction: The team identified a NCV of 10 CFR Part 50 Appendix B, Criterion V for the licensee's failure to provide documented instructions appropriate to the circumstances for the remote visual examination of the final PT examination completed on repaired nozzle No. 61. Specifically, OI 03-1240857-006 "BWOG CRDM Nozzle Top Down Inspection Tooling Operating Instructions," did not include guidance for control of spacer sizes or camera field of view necessary to ensure that the entire examination surface area was viewed.

Description: On April 13, 2010, the team identified that the licensee failed to use a procedure with adequate controls to ensure flaws were detected in the CRDM nozzle repairs.

After completion of the post-weld machining of the CRDM nozzle No. 61, the licensee's vendor completed the remote visual examination of the final PT acceptance examination for the repaired surface of the vessel head penetration in accordance with OI 03-1240857-006 "BWOG CRDM Nozzle Top Down Inspection Tooling Operating Instructions." Task 32 of OI 03-1240857-006, required in part "to insert the required number of spacers under the pointing collar (to ensure overlap)." However, the team identified that the portions of the final acceptance PT examination on the post repaired surface of nozzle No. 61 had not been viewed because this procedure lacked specific instructions to control the size of spacers used to establish the vertical position of the remote camera. Additionally, this procedure lacked guidance to establish a specific field of view for the remote camera. Subsequently, the licensee's vendor estimated that approximately 14 percent of the examination area was not viewed. The team was concerned that failure to view the entire examination surface could have resulted in failure to identify fabrication flaws such as repair weld cracks.

The licensee had not reviewed nor approved the vendor Procedure OI 03-1240857-006 because the licensee relied on the vendor's quality assurance program for the repair process operating instructions. To correct this issue, the licensee's vendor Revised OI 03-1240857-006 to provide additional instructions to ensure complete examination coverage with the remote camera system. The licensee subsequently approved the vendor's corrective actions (vendor CR 2010-3544) that revised OI 03-1240857-006 and repeated PT examinations on nozzle No. 61 and nine additional nozzles with incomplete coverage.

Analysis: The team determined that the licensee's failure to use documented instructions appropriate to the circumstances for the detection of flaws in the repaired nozzles was a performance deficiency that impacted the Initiating Events Cornerstone attribute of Equipment Performance. The team evaluated this performance deficiency against the examples of minor and more than minor findings identified in IMC 0612, Appendix E and determined that there were no sufficiently similar examples. Specifically, this was not considered a work in progress deficiency, because the licensee had completed the PT nozzle examinations using the ineffective visual examination procedure and no other quality assurance measure existed to detect this error prior to placing the affected nozzles in service.

The team determined that this finding was more than minor because if left uncorrected, the failure to use an adequate procedure for detecting flaws could become a more significant safety concern. Absent NRC identification, the licensee would not have examined the entire surface of the repaired nozzle No. 61 and nine other nozzles that could have allowed cracks to go undetected. Undetected cracks returned to service in the repair welds would place the RVCH at increased risk for through-wall leakage and/or nozzle failure. Therefore, this finding adversely affected the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions.

The team completed a significance determination, in accordance with Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 0609.04, "Phase I - Initial Screening and Characterization of findings," Table 4a for the Initiating Events Cornerstone. The issue was corrected promptly, weld cracks were not returned to service, and the team answered "no" to the Phase I screening question that asked assuming the worst case degradation would the finding result in exceeding the Technical Specification limit for any reactor coolant system leakage. Therefore, the finding screened as having very low safety significance (Green).

This finding had a cross-cutting aspect in the area of Human Performance, Work Practices per IMC 0310 (Item H.4(c)) because the licensee did not provide adequate supervisory and management oversight of work activities including contractors such that nuclear safety was supported. Specifically, the licensee failed to provide an adequate oversight in that no licensee review was completed for the inadequate vendor Procedure OI 03-1240857-006. The team concluded that this was the primary cause of the finding based upon discussions with licensee and vendor staff.

Enforcement: Title 10 CFR Part 50 Appendix B, Criterion V "Instructions, Procedures, and Drawings" required in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances.

Contrary to the above, as of April 13, 2010, the licensee had not provided documented instructions appropriate to the circumstances for the remote visual examination of the final PT examination completed on repaired nozzle No. 61. Specifically, the instructions in OI 03-1240857-006 “BWOG CRDM Nozzle Top Down Inspection Tooling Operating Instructions,” did not include guidance for control of spacer sizes or camera field of view necessary to ensure that the entire examination surface area was viewed. Failure to use a procedure with instructions appropriate to the circumstances for examinations of repaired CRDM nozzles is a violation of 10 CFR Part 50 Appendix B, Criterion V. Because of the very low safety significance of this finding and because the issue was entered into the licensee’s Corrective Action Program (CR 10-77201), it is being treated as a NCV, consistent with Section VI.A.1 of the Enforcement Policy (November 28, 2008) (NCV 05000346/2010008-03).

b.5 Abrasive Water Jet Remediation (AWJR) of CRDM Nozzle Step

In the final repair step, the licensee applied an AWJR process to the inside wetted surface of the expanded portions of the CRDM nozzle susceptible to PWSCC. This process utilized high-pressure water with entrained abrasive material that removed a small amount of nozzle material while imposing a compressive residual stress on the inside nozzle surface. This step was necessary because the hydraulic expansion step increased the residual tensile stress in the nozzle material and without remediation it may increase the material’s susceptibility to PWSCC. The team observed portions of the AWJR process and no deviations from the work instructions were observed. The licensee’s vendor experienced minor equipment problems associated with the abrasive feed mechanism that slowed work progress, but did not affect successful completion of this step.

Because of the team’s finding related to the inadequate procedure for viewing the post repair PT examinations, the licensee repeated the PT examination of nozzle No. 58 following the AWJR step. During this PT examination a surface flaw indication was detected that had not been identified during the post AWJR visual examination. The licensee believed that the AWJR process likely removed surface material and opened up this small indication. The licensee repaired this shallow defect by weld buildup followed by grinding. The licensee performed post repair PT and UT examinations to confirm that the flaw was removed and the nozzle was acceptable for service. Based on the failure of the post AWJR visual examination to detect the indication in nozzle No. 58, the licensee performed PT examinations following AWJR on each of the repaired nozzles. Following these additional PT examinations, the licensee detected a surface flaw at nozzle No. 4 which was then repaired by weld buildup and grinding. The licensee performed post repair PT and UT examinations to confirm that the flaw was removed and the nozzle was acceptable for service.

4OA6 Meetings

.1 Exit Meeting

On September 9, 2010, the team presented the inspection results to Mr. Barry Allen and members of the licensee staff at a public exit meeting held at Oak Harbor, Ohio (NRC presentation materials and list of attendees-ADAMS Accession Nos. ML102310165, ML102510346, and ML102861816). This constitutes the public meeting summary for the exit meeting. The team reviewed proprietary documents during this inspection and asked the licensee to identify any report input material that was considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Allen, Site Vice President
B. Boles, Director, Site Operations
V. Kaminskas, Director, Site Engineering
G. Wolf, Regulatory Compliance Supervisor
K. Byrd, Manager, Design Engineering
A. Bless, Regulatory Compliance
K. Spencer, Regulatory Compliance
K. Zellers, Design Engineering

Nuclear Regulatory Commission

J. Rutkowski, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000346/2010-008-01	NCV	Unqualified PT Procedure For CRDM Nozzle Repair Welds
05000346/2010-008-02	NCV	Unqualified Weld Repair Applied For CRDM Nozzle No. 4
05000346/2010-008-03	NCV	Inadequate Procedure For Viewing of Remote PT on Nozzle No. 61 Repair Welds

Closed

05000346/2010-002-00	LER	Control Rod Drive Nozzle Primary Water Stress Corrosion Cracking and Pressure Boundary Leakage

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the team reviewed the documents in their entirety but rather that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

Calculations and Evaluations

Structural Integrity Associates, Document 1000422.401; Crack Growth Evaluations of CRDM Penetration Nozzles at Davis-Besse Nuclear Power Station; dated April 28, 2010

Structural Integrity Associates, Document 1000422.401; Crack Growth Evaluations of CRDM Penetration Nozzles at Davis-Besse Nuclear Power Station; dated June 24, 2010

Calculation C-ME-099.99-013, Effective Degradation Years (EDY) Calculation for Alloy 600/ 82/ 182 PWSCC Susceptibility Determination Related Applications" Revision 3

Calculation C-3223-00-03; Davis-Besse Original and Replacement Head CRDM Nozzle Welding Residual Stress Comparison; Revision 0

Corrective Action Program Documents As a Result of NRC Inspection

CR 10-77957; Repair weld on CRDM nozzle 4 did not comply with Code Case; dated June 8, 2010

CR10-76773; IDBT Weld Repair Process/Discrepancy between OI and WPS Parameters; May 11, 2010

CR10-74059; Implementation of ASME Code Case N-729-1 for RVCH Examinations, dated March 23, 2010

CR 10-75187; PT Indications Were Found on Nozzles 55 and 67; Dated April 10, 2010

CR 10-75709; NRC Questions on AREVA Liquid Penetrant Procedure 54-ISI-244-10; Dated April 16, 2010

CR 10-76471; CRDM PT Decision Tree Acceptance Criteria; May 6, 2010

R 10-76029; Reactor Vessel Head CRDM Nozzle to CRDM Adapter Butt Weld Inspection; dated April 28, 2010

CR 10-77201; CRDM Nozzle PT Exams Not Completed; dated May 22, 2010

CR 10-75279; Questions about the AREVA Top down PT Tool Training; dated April 10, 2010

CR 10-78725; NRC Questions of AREVA Visual Examination Report; dated June 23, 2010

Corrective Action Program Documents - Licensee

CR 10-73323; CRDM Nozzle and Weld Cracking, Pressure Boundary Leakage; March 12, 2010

CR10-76511; DBRV: Two Rejectable Indications Identified During PT Exam on Nozzle 60: May 7, 2010

CR10-76234; 16RFO DBRV-Indication on Nozzle 3, J-Groove Weld/Clad Interface; May 1, 2010

CR10-76205; 16RFO DBRV-Eddy Current Indication at Nozzle 27 J-Groove Weld; April 30, 2010

CR10-76233; 16RFO DBRV-Eddy Current Indication at Nozzle 1 J-Groove Weld; May 1, 2010

CR10-76209; 16RFO DBRV-Eddy Current Indication at Nozzle 29; April 30, 2010

CR10-75709; NRC Questions on AREVA Liquid Penetrant (PT) Procedure 54-ISI-244-10; April 21, 2010

CR10-76210; 16RFO – Eddy Current Indication Nozzle 60 J-Groove Weld; April 30, 2010

CR10-76212; 16RFO – Eddy Current Indication Nozzle 66 J-Groove Weld; April 30, 2010

CR10-76029; RPV Head CRDM Nozzle No.14 to CRDM Adapter Buttweld; April 28, 2010

CR10-76236; 16RFO – Eddy Current Indication Nozzle 40 J-Groove Weld; May 1, 2010

CR10-76235; 16RFO – Eddy Current Indication Nozzle 21 J-Groove Weld; May 1, 2010

CR10-76505; Liquid Penetrant (PT) Indications Detected on CRDM No.53; May 6, 2010

CR10-75467; Nozzle 33 Rejectable Indication Identified During the Post Machining PT Exam; April 16, 2010

CR10-75670; Recordable Indication on Reactor Head Nozzle No. 43; April 21, 2010

CR10-76907; DNVR: Nozzle No. 4 Questionable Indications; May 14, 2010

CR10-76888; DBRV: Nozzle No. 4 Weldability Issues; May 13, 2010

CR10-76585; DBRV: Nozzle No. 60 PT Indications; May 6, 2010

CR10-77201; DBRV: CRDM Nozzle PT Exams Not Complete; May 17, 2010

CR10-75552; DBVR: 2nd Layer of Nozzle No. 4 Weld Will Not Fuse to Boric Acid Contamination; April 18, 2010

CR10-73323; CRDM Nozzle and Weld Cracking with Pressure Boundary Leakage; March 12, 2010

CR10-77539; RV Head repair – Failed PT on CRD Nozzle No. 4; May 30, 2010

CR10-77540; RV Head Repair – failed Post AWJ of CRDM Nozzles, Nos. 3, 48, 57; May 30, 2010

CR10-77541; Failed PT on CRD Nozzle No. 10; May 30, 2010

CR10-77550; Failed Pre-Weld PT on Nozzle No. 4; May 31, 2010

CR10-77564; CRD Nozzle No. 1 Rejected PT; June 1, 2010

CR10-75027; DB Reactor Vessel Head AREVA CR 2010-2374; April 8, 2010

CR10-75030; DB reactor Vessel Head repair AREVA CR 10-2375; April 8, 2010

CR10-75163; BDRV: Nozzle No. 4 Rework for Undercut of Bevel (ACR 2010-2431); April 9, 2010

CR10-77346; DBRV; Nozzle No. 58 Rejectable Post-Weld PT; May 25, 2010

Corrective Action Records – Vendor - AREVA

2010-2460 CR Process; Two PT Indications Were Found in the Machine Region of the Penetration No. 55; April 11, 2010

2010-2467 CR Process; One PT Indication Was Found In the Machine Region of Penetration No. 67; April 11, 2010

2010-3275 CR Process; Excessive Bleed Out During PT of Nozzle No.4; May 11, 2005

2010-3544 CR Process; PT Exam on Nozzle 61 had Missed Coverage; May 22, 2010

2010-3341-CR Process; Unable to get a Successful Weld in RX Head Nozzles No.4; May 13, 2010

2010-3170-CR Process; RVH Nozzle 60, Post Machining PT of Weld Prep had Indications in the Area of Interest; May 6, 2010

2010-3330-CR Process; IDTB Weld Repair Process/Discrepancy between OI and WPS Parameters; May 12, 2010

2010-3734 CR Process; 2 Rejectable PT Indications, Reactor Head Nozzle No. 1; June 1, 2010

2010-3723 CR Process; A Visual Anomaly was noted during the Post Water Jet (AWJ) Visual Exam of Nozzle 4; May 29, 2010

2010-3644 CR Process; Post AWJ PT Indications Nozzle 58; May 25, 2010

Drawings

7749-M-197-2-X-4; Mirror Insulation Reactor Vessel dated October 4, 2004

WE No. 550182; Grooveman Cal-Check Block; Revision A

AREVA No. 029134305E; Davis-Besse CRDM Nozzle ID Temper Bead Weld Repair; Revision 4

Framatome ANP No. 142179; Closure Head Sub-Assembly; Revision 10

Framatome ANP No. 142182; Closure Head Assembly; Revision 8

Framatome ANP No. 154613; Arrangement Reactor Vessel Long Section; Revision 8

Framatome ANP No. 154614; Arrangement Reactor Vessel Sections; Revision 8

550182; Grooveman Cal-Check Block; Revision A

02-9134305E; Davis- Besse CRDM Nozzle ID Temper Bead Weld Repair: Revision 2

02-9134305E; Davis- Besse CRDM Nozzle ID Temper Bead Weld Repair: Revision 3

02-9134305E; Davis- Besse CRDM Nozzle ID Temper Bead Weld Repair: Revision 4

5015339E; CRDM Nozzle ID Temperbead Weld Repair; Revision 1

50-9134306; Appendix B – repair of Nozzle No.4 CR 2010-2644; April 27, 2010

4500096961; Grooveman Cal-Check Block; September 13, 2002

02-8041044C; RVCH Repair Mockup NDE Flaw Specifications and Locations Davis-Besse Unit 1; Revision 1

Non-Destructive Examination Reports

Eddy Current Report Sheet BDAV1-R16-GM01-01-01; Nozzle 01; dated May 1, 2010

Eddy Current Report Sheet BDAV1-R16-GM01-03-01; Nozzle 03; dated May 1, 2010

Eddy Current Report Sheet BDAV1-R16-GM01-21-01; Nozzle 21; dated May 1, 2010

Eddy Current Report Sheet BDAV1-R16-GM01-27-01; Nozzle 27; dated May 1, 2010

Eddy Current Report Sheet BDAV1-R16-GM01-29-01; Nozzle 29; dated May 1, 2010

Eddy Current Report Sheet BDAV1-R16-GM01-40-01; Nozzle 40; dated May 1, 2010

Eddy Current Report Sheet BDAV1-R16-GM01-60-01; Nozzle 60; dated May 1, 2010

Eddy Current Report Sheet BDAV1-R16-GM01-66-01; Nozzle 66; dated May 1, 2010

WDI-PJF-1304903-FSR-001; Reactor Vessel Head Inspection Final Report (Eddy Current); Spring, 2010; Revision 0

Liquid Penetrant Examination Report 16-PT-049; CRDM Nozzle 57; dated April 1, 2010

Liquid Penetrant Examination Report 16-PT-050; CRDM Nozzle 40; dated April 1, 2010

Liquid Penetrant Examination Report 16-PT-051; CRDM Nozzle 64; dated April 1, 2010

Liquid Penetrant Examination Report 16-PT-052; CRDM Nozzle 60; dated April 1, 2010

Liquid Penetrant Examination Report 16-PT-053; CRDM Nozzle 42; dated April 1, 2010

Liquid Penetrant Examination Report 16-PT-054; CRDM Nozzle 66; dated April 1, 2010

Liquid Penetrant Examination Report 16-PT-055; CRDM Nozzle 16; dated April 1, 2010

Liquid Penetrant Examination Report 16-PT-056; CRDM Nozzle 11; dated April 1, 2010

Liquid Penetrant Examination Report 16-PT-057; CRDM Nozzle 48; dated April 1, 2010

Liquid Penetrant Examination Report 16-PT-058; CRDM Nozzle 12; dated April 1, 2010

Liquid Penetrant Examination Report 16-PT-059; CRDM Nozzle 53; dated April 1, 2010

Liquid Penetrant Examination Report 16-PT-075; CRDM Nozzle 53; dated May 10, 2010

Liquid Penetrant Examination Report NOZ 33-NDE-200-00; CRDM Nozzle No.33; April 14, 2010.

Liquid Penetrant Examination Report NOZ 33-NDE-200-00; CRDM Nozzle No.33; April 15, 2010

Liquid Penetrant Examination Report NOZ 33-NDE-200-00; CRDM Nozzle No.33; April 24, 2010

Liquid Penetrant Examination Report NOZ 67-NDE-200-00; CRDM Nozzle No.67; April 10, 2010

Liquid Penetrant Examination Report NOZ 04-NDE-200-00; CRDM Nozzle No.04; April 15, 2010

Liquid Penetrant Examination Report NOZ 28-NDE-200-00; CRDM Nozzle No.28; April 14, 2010

Liquid Penetrant Examination Report NOZ 59-NDE-200-00; CRDM Nozzle No.59; April 15, 2010

Liquid Penetrant Examination Report NOZ 61-NDE-200-00; CRDM Nozzle No.61; April 14, 2010

Liquid Penetrant Examination Report NOZ 43-NDE-200-00; CRDM Nozzle No.43; April 14, 2010

Liquid Penetrant Examination Report NOZ 55-NDE-200-00; CRDM Nozzle No.55; April 10, 2010

Liquid Penetrant Examination Report NOZ 24-NDE-200-00; CRDM Nozzle No.24; April 14, 2010

Liquid Penetrant Examination Report NOZ 51-NDE-200-00; CRDM Nozzle No.51; April 12, 2010

Liquid Penetrant Examination Report NOZ 10-NDE-200-00; CRDM Nozzle No.10; April 13, 2010

Liquid Penetrant Examination Report NOZ 58-NDE-200-00; CRDM Nozzle No.58; April 12, 2010

Liquid Penetrant Examination Report NOZ 58-NDE-200-01; CRDM Nozzle No.58; April 15, 2010

Liquid Penetrant Examination Report NOZ 43-NDE-320-00; PT of CRDM Nozzle No.43; April 21, 2010

Liquid Penetrant Examination Report NOZ 60-NDE-200-00; PT of CRDM Nozzle No.60 (Sequence 200); May 6, 2010

Liquid Penetrant Examination Report NOZ 1-NDE-B10-00; CRDM Nozzle No.1; May 31, 2010

Liquid Penetrant Examination Report NOZ 3-NDE-B10-00; CRDM Nozzle No.3; May 31, 2010

Liquid Penetrant Examination Report NOZ 4-NDE-B10-00; CRDM Nozzle No.4; May 29, 2010

Liquid Penetrant Examination Report NOZ 4-NDE-C40-00; CRDM Nozzle No.4; May 30, 2010

Liquid Penetrant Examination Report NOZ 4-NDE-C130-00; CRDM Nozzle No.4; May 31, 2010

Liquid Penetrant Examination Report NOZ 10-NDE-B10-00; CRDM Nozzle No.10; May 30, 2010

Liquid Penetrant Examination Report NOZ 16-NDE-B10-00; CRDM Nozzle No.16; May 31, 2010

Liquid Penetrant Examination Report NOZ 21-NDE-B10-00; CRDM Nozzle No.21; May 30, 2010

Liquid Penetrant Examination Report NOZ 24-NDE-B10-00; CRDM Nozzle No.24; May 31, 2010

Liquid Penetrant Examination Report NOZ 27-NDE-B10-00; CRDM Nozzle No.27; May 31, 2010

Liquid Penetrant Examination Report NOZ 28-NDE-B10-00; CRDM Nozzle No.28: May 31, 2010

Liquid Penetrant Examination Report NOZ 29-NDE-B10-00; CRDM Nozzle No.29; May 30, 2010

Liquid Penetrant Examination Report NOZ 33-NDE-B10-00; CRDM Nozzle No.33; June 1, 2010

Liquid Penetrant Examination Report NOZ 40-NDE-B10-00; CRDM Nozzle No.40; May 30, 2010

Liquid Penetrant Examination Report NOZ 43-NDE-B10-00; CRDM Nozzle No.43; May 29, 2010.

Liquid Penetrant Examination Report NOZ 51-NDE-320-01; CRDM Nozzle No.51; May 26, 2010

Liquid Penetrant Examination Report NOZ 55-NDE-320-01; CRDM Nozzle No.55: May 26, 2010.

Liquid Penetrant Examination Report NOZ 57-NDE-B10-00; CRDM Nozzle No.57; May 29, 2010

Liquid Penetrant Examination Report NOZ 58-NDE-320-01; CRDM Nozzle No.58; May 25, 2010.

Liquid Penetrant Examination Report NOZ 58-NDE-320-02; CRDM Nozzle No.58; May 27, 2010

Liquid Penetrant Examination Report NOZ 58-NDE-D40-00; CRDM Nozzle No.58; May 28, 2010

Liquid Penetrant Examination Report NOZ 58-NDE-D130-00; CRDM Nozzle No.58; May 28, 2010

Liquid Penetrant Examination Report NOZ 59-NDE-320-01; CRDM Nozzle No.59; May 29, 2010

Liquid Penetrant Examination Report NOZ 60-NDE-B10-00; CRDM Nozzle No.60; May 29, 2010

Liquid Penetrant Examination Report NOZ 61-NDE-320-01; CRDM Nozzle No.61; May 27, 2010

Liquid Penetrant Examination Report NOZ 64-NDE-B10-00; CRDM Nozzle No.64; May 29, 2010

Liquid Penetrant Examination Report NOZ 66-NDE-B10-00; CRDM Nozzle No.66; May 30, 2010

Liquid Penetrant Examination Report NOZ 67-NDE-320-01; CRDM Nozzle No.67: May 27, 2010

Eddy Current Examination Report - WDI-PJF-1304903-FSR-001 Reactor Vessel Head Inspection Final Report Spring, 2010, Revision 0

Ultrasonic Examination Report - AREVA Document 51-9135976-000; Davis-Besse Unit 1, RFO 16, Reactor Head Inspection Report; dated May 27, 2010

VT-1 Visual Examination Data Sheet RX Vessel Head Noz-4-DB-001; May 12, 2010

VT-1 Visual Examination Data Sheet; NOZ 4-NDE-C125-01; dated May 23, 2010

Other Documents

ECP 10-0141-000; CRDM Nozzle Repair; Revision 0

WO52-5465; Inspection Report for Grooveman Cal-Check Block; September 25, 2002

6012737 B-1: No-Go Gauge (Machining); Revision 1

6007422B; CRDM Repair-Grinding Process No-Go Gauge Guide; Revision 2

First Energy Intra-Company Memorandum; Review of Contracted NDE Personnel Certifications Who are Working Under Their QA Program-Wesdyne Memo No. 2; dated April 28, 2010

First Energy Intra-Company Memorandum; Review of Contracted NDE Personnel Certifications Who are Working Under Their QA Program-Wesdyne; dated April 13, 2010

First Energy Intra-Company Memorandum; Acceptance of Westinghouse Non-Destructive Examination Procedures-16RFO Grooveman Eddy Current; dated April 26, 2010

Wesdyne Letter WDI-LTR-QA-10-25; NDE Personnel Certification Training”; dated April 8, 2010

MRP-103; Reactor Vessel Head Nozzle and Weld Safety Assessment for B &W Plants; dated April 2004

PNNL Report 19362; Evaluation of Ultrasonic Time-of-Flight Diffraction Data for Selected Control Rod Drive Nozzles from Davis-Besse Nuclear Power Plant; dated April 2010

MRP 110; Reactor Vessel Closure Head Penetration Safety Assessment for U.S. PWR Plants; dated April 2004

Memorandum from Jack R. Strosnider, Director Division of Engineering to Samuel J. Collins, Director Office of NRR, Results of Independent Evaluation of Recent Reactor Vessel Head Penetration Cracking; dated September 7, 2001

CR 10-77079; Nozzle 4 Unacceptable Condition around Triple Point; May 14, 2010

EC Calibration Data Sheet; For Procedure WDI-ET-002 Revision 13; April 27, 2010

TSS-10-00056; Acceptance of Westinghouse Non-Destructive Examination Procedure – 16 RFO Grooveman Eddy Current; April 23, 2010

WDI-T-J-1028; ASME Section V, Article 14, Technical Justification for Eddy Current Inspections of RVH; Revision 0

WDI-TJ-1008; Evaluation of the Effect of Increasing RVHI RF Data Cable Length to 75 Feet; Revision 0

WDI-TJ-002-02; Technical Justification for Eddy Current Testing of J-Groove Welds at CRDM Penetrations Using Procedure ISI-ET-001; "Eddy Current Inspection of J-Groove Welds in Vessel Head Penetrations"; Revision 0

WesDyne Procedure WDI-ET-002; "IntaSpect Eddy Current Inspection of J-Groove Welds in Vessel Head Penetrations"; Revision 0

WDI-ET-004; IntraSpect Eddy Current Analysis Guidelines; Revision 14

ATR-1075; IntraSpect Eddy Current Analysis Guidelines Inspection of Reactor Vessel Head Penetrations; Revision A

WDI-TJ-027-04; Evaluation of the Ability for ET Auto Analysis to detect Liftoff in Grooveman Data; Revision 0

WDI-LTR-QA-10-25; NDE Personnel Certification Transmittal; April 8, 2010

Intra-Company Memo; Review of Contracted NDE Personnel Certifications who are Working Under their QA Program – WesDyne; April 12, 2010

Intra-Company Memo; Review of Contracted NDE Personnel Certifications Who are Working Under their QA Program – WesDyne Memo No.2; April 28, 2010

BDAV1-R16-GM01-53-01; Penetration No.53; April 29, 2010

NA-QC-00191; Liquid Penetrant Examination; Revision 6

FLUS Operating Manual, Section 7.1 through 7.4; No date

FLUS System Output Data, Cycle 15 and 16; Channel 1 and 2 Data; No Date

WCAP-16423-NP; Pressurized Water Reactor Owners Group Standard Process and Methods for Calculating RCS Leak Rate for Pressurized Water Reactors; September 2006

Davis-Besse Program Manual; RCS Integrated Leakage Program; Revision 1

Davis-Besse Excel Data Bases; Cycle 15 Leakage Trending, No Date

Davis-Besse Excel Data Bases; Cycle 16 Leakage Trending, No Date

DB-OP-1200, Attachment 7 Results; RCS Boundary Valve Leakage Checklist; November 29, 2008, and February 10, 2009

DB-SC-4002; FLUS Leak Detection System Functional Test; Revision 5

DB-PF-10140; FLUS Leak Detection System Acceptance Test; Revision 3

DB-SC-5002; FLUS System Changes; Revision 1

EN-DP-1171; Engineering Implementation of the RCS Integrated Leakage Program; Revision 2

ECR 02-0792-00; Containment Leak Detection System (FLUS) Installation; Revision 2

Westinghouse document RTU-MCE-10-36; Final Report-Summary of Davis-Besse Unit 1 CRDM Nozzle Boat Samples Destructive Examinations; dated May 2010.

FENOC Letter L-10-099, 10 CFR 50.55a Request for Alternate Repair Methods for Reactor Pressure Vessel Head Penetration Nozzles (ML100960276); dated April 1, 2010

FENOC Letter L-10-143, Request for Additional Information Response and Supplement to 10 CFR 50.55a Request RR-A34 For Alternate Repair Methods for Reactor Pressure Vessel Penetration Nozzles (ML101400402); dated May 17, 2010

NRC Letter Davis-Besse Nuclear Power Station, Unit No. 1 – Summary of Teleconference RE: Verbal Authorization for Relief Request RR-A34 (ML1016001470), dated June 14, 2010

Work Order 200419786; Grind and PT Penetration 53; dated June 11, 2010

AREVA Memo; Additional VT Training in Accordance with Code Case N-638-4; dated May 11, 2010

32-5012424-03; CRDM Temper Bead Bore Weld Analysis; dated August 6, 2001

N-729-1; Alternative Examination Requirements for PWR Rector Vessel Upper Heads With Nozzles having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1; dated March 28, 2006

N-686; Alternative Requirements for Visual Examinations, VT-1, VT-2, and VT-3 Section XI, Division 1; dated February 14, 2003

N-638-1; Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique Section XI, Division 1; dated February 13, 2003

N-638-4; Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique Section XI, Division 1; dated October 5, 2006

03-1240857-05; Personnel Training Roster (PTR) BWOG CRDM Nozzle Top Down Inspection Tooling Operating Instructions (PT Operations Only); dated April 9, 2010

54-PQ-244-06; Procedure Qualification to Establish Adequate Lighting Requirements for the Examination as Stated in 54-ISI-244-10; dated April 11, 2010

30-9136057-000; SDCN for 54-ISI-244-10; dated April 10, 2010

30-9136059-000; SDCN for 54-ISI-244-10; dated April 10, 2010

L-10-099; 10CFR 50.55a Request for Alternate Repair Methods for Reactor Pressure Vessel Head Penetration Nozzles; dated April 1, 2010

1006800; EPRI – Welding on Materials Exposed to Boric Acid; dated June 2003

MTSD; Tapmatic AquaCut Cutting Fluid; Revision 3

Sample T29.1.N100911107.000; Trace Constituent Analysis Lab Report, Tapmatic Aqua Cutting Fluid; dated March 23, 2010

MRSR-DB-4532; Trace Contaminant Analysis Report for Oil of Wintergreen; dated May 7, 2010

51-9136307-001; Evaluation of Davis-Besse Boric Acid Deposits from RFO-16 CRDM Nozzle Inspections; dated March 29, 2010

51-9137763-000: Davis-Besse RFO16 CRDM “J” Groove Weld Fusion Zone Indication Survey; dated May 17, 2010

51-9136423-001; Analysis of Davis-Besse Boric Acid Deposits from RFO-16 CRDM Nozzle Inspections; dated April 29, 2010

51-5016342-005; Ambient Weld Interpass Temperature Evaluation; dated September 28, 2006

51-9136137-000; Weld Interpass Temperature Monitoring for CRDM IDTB Repairs; dated April13, 2010

51-9136423-000; Analysis of Davis-Besse Boric Acid Deposits from RFO-16 CRDM Nozzle Inspections; dated April 16, 2010

51 - 9137401 – 000; Evaluation of Fluid Temperature in DB RV Closure Head; dated May 19, 2010

Work Order No. 310766; AREVA NP, Inc. Aluminum Test Block; dated April 20, 2010

Reedy Engineering Letter; CR10-76505, ASME Section XI Case N-729-1, Evaluation of PT indications; dated May 20, 2010

55111195; Crack Growth Evaluation of Control Rod Drive Mechanism Nozzle Penetration at Davis-Besse Nuclear Power Station; dated April 28, 2010

Procedures

AREVA Procedure Number 54-ISI-244-10; Nondestructive Examination Procedure Liquid Penetrant Examination of Reactor Vessel Head Penetrations from the Inside Surface; dated August 13, 2007

DB-SP-3357; RCS Water Inventory Balance; Revision 15

DB-OP-6418; FLUS Containment leak Detection System; Revision 2

DB-OP-1200; RCS Management; Revision 12

54-ISI-244-10; Liquid Penetrant Examination of Reactor Vessel Head Penetrations from the Inside Surface; Revision 10

54-ISI-244-12; Liquid Penetrant Examination of Reactor Vessel Head Penetrations from the Inside Surface; Revision 12

54-ISI-245-01; Color Contrast Water Washable Liquid Penetrant Examination Procedure; Revision 1

NA-QC-00191; Liquid Penetration Examination; Revision 6

54-ISI-365-02; Visual Inspection of Pressure Vessel Internals, Attachments and Internal Surfaces; Revision 2

54-ISI-491-07; Multi-Frequency Rotating Eddy Current Examination of Reactor Vessel Head Penetrations; October 2, 2006

54-ISI-178-07; Ultrasonic Examination of Temperbead Weld Repairs on PWR Upper Head Nozzles and BWR Lower Head Housings; Revision 7

55-OI0031-012; Ambient I.D.T.B. Welding CRDM Penetration of Reactor Vessel Closure Heads Utilizing Modified Local Cavity Weldhead; Revision 12

03-1240857-006; Davis-Besse ID PT Tooling Operating Instructions; Revision 6

03-8041779; Operating Instruction, Modified 212B, IDTB Bottom-Up Repair, Davis-Besse (Appendix B – Top Down Hard Stop Setup Calculation); Revision 3

03-5014699; Waterjet Remediation Operating Instruction Top-Down Configuration for B&W Design Plants; Revision 4

WDI-ET-002; IntraSpect Eddy Current Inspection of Vessel Head Penetration J-Welds and Tube OD Surfaces; Revision 13

Qualification and Certification Records

EPRI Performance Demonstration Qualification Sheet No 1021; ID 6710838; dated March 8, 2010

EPRI Performance Demonstration Qualification Sheet No 1020; ID 4982915; dated March 8, 2010

EPRI Performance Demonstration Qualification Sheet No 1022; ID 9022381; dated March 8, 2010

EPRI Performance Demonstration Qualification Sheet No 1023; ID 7717081; dated March 8, 2010

EPRI Performance Demonstration Qualification Sheet No 1024; ID 5135180; dated March 8, 2010

AREVA Certificate of Personnel Qualification; ID C5341; dated March 9, 2010

AREVA Certificate of Personnel Qualification; ID K2915; dated March 9, 2010

AREVA Certificate of Personnel Qualification; ID S2381; dated March 9, 2010

AREVA Certificate of Personnel Qualification; ID S7081; dated March 9, 2010

AREVA Certificate of Personnel Qualification; ID T5180; dated March 9, 2010

AREVA Certificate of Personnel Qualification; ID 1704956330131; dated March 10, 2010

AREVA Certificate of Personnel Qualification; ID O9426; dated October 16, 2009

AREVA Certificate of Personnel Qualification; ID P6274; dated February 8, 2010

AREVA Certificate of Personnel Qualification; ID W4087; dated January 26, 2010

AREVA Certificate of Personnel Qualification; ID L6868; dated January 22, 2010

AREVA Certificate of Personnel Qualification; ID S2381; dated March 9, 2010

AREVA Certificate of Personnel Qualification; ID J6276; dated March 13, 2007

AREVA Certificate of Personnel Qualification; ID R6452; dated August 4, 2008

Liquid Penetrant Procedure Qualification – Procedure 54-ISI-244-10; dated April 11, 2010

Liquid Penetrant Procedure Qualification – Procedure 54-ISI-244-10; dated April 22, 2010

Video Records

Nozzle No.61 PT (Pre-weld); April 14, 2010

Nozzle No.61; PT (Post-Weld); April 23, 2010

Nozzle No.4: 54-ISI: 365-02; May 11, 2010

Nozzle No.4: Weld Related; May 13, 2010

Nozzle No.10; PT (Pre-weld); Sequence 200; April 14, 2010

Nozzle No.43, No.24, No.33; PT; April 14, 2010

Nozzle No.59; PT (Pre-weld) Sequence 200; April 14, 2010

Nozzle No.33; PT; April 16, 2010

Nozzle No.58; Post Grind, Seq. 200; April 15, 2010

Nozzle No.67; Post Etching; April 11, 2010

Nozzle No.67; PT; April 13, 2010

Nozzle No.4; PT Exam; April 15, 2010

Nozzle No.24; Weld Prep PT; April 14, 2010

Nozzle No.33; VT-1; April 16, 2010

Nozzle No.61; VT-1 Informational; April 16, 2010

Nozzle No.4; Welding Triple Point; April 19, 2010

Nozzle No.4; VT-1; April 19, 2010

Nozzle No.28; PT (Pre-weld); April 14, 2010

Nozzle No.51; PT Exam; April 22, 2010

Nozzle No.59; Post-weld; April 23, 2010

Nozzle No.67; Sequence 320; April 23, 2010

Nozzle No.33; PT; April 21, 2010

Nozzle No.58; PT exam; April 21, 2010

Nozzle No.58; PT exam; April 21, 2010

Nozzle No.33; PT (Pre-weld) Seq. 200; April 24, 2010

Nozzle No.33; PT (Info only); April 24, 2010

Nozzle No.61 PT (Pre-weld); April 23, 2010

Nozzle No.55 PT (Pre-weld); April 21, 2010

Nozzle No.21; PT (Pre-Weld) Sequence 200; May 6, 2010

Nozzle No.29; PT (Pre-weld) Sequence 200; May 6, 2010

Nozzle No. 27; PT (Pre-weld) Sequence 200; May 6, 2010

Nozzle No.60; PT (pre-weld) Sequence 200; May 6, 2010

Nozzle No.3; PT (Pre-weld) Sequence 200; May 6, 2010

Nozzle No.66; PT (Pre-weld) Sequence 200; May 6, 2010

Nozzle No.40; PT (Pre-weld) Sequence 200; May 6, 2010

Nozzle No.16; PT (Pre-weld) Sequence 200; May 17, 2010

Nozzle No.48; Seq. 200 TDPT; May 7, 2010

Nozzle No.57; Seq. 200 TDPT; May 7, 2010

Nozzle No.1; Seq. 200 TDPT; May 6, 2010

Nozzle No.64; PT (Pre-weld) Sequence 200; May 7, 2010

Nozzle No.4; 54-ISI: 365-02; May 11, 2010

Nozzle No.1; PT (Post-weld) Sequence 320; May 16, 2010

Nozzle No.64; PT (Post-weld) Sequence 320; May 16, 2010

Nozzle No.29; Sequence. 320 TDPT; May 16, 2010

Nozzle No.40; PT (Post-weld) Sequence 320; May 16, 2010

Nozzle No.66; PT (Post-weld) Sequence 320; May 16, 2010

Nozzle No.27; PT (Post-weld) Sequence 320; May 16, 2010

Nozzle No.57; PT (Post-weld) Sequence 320; May 17, 2010

Nozzle No.16; PT (Post-weld) Sequence 320; May 17, 2010

Nozzle No.60; PT (Post-weld) Sequence 320; May 17, 2010

Nozzle No.3; PT (Post-weld) Sequence 320; May 17, 2010

Nozzle No.21; PT (Post-weld) Sequence 320; May 17, 2010

Nozzle No.33 PT (Post-Weld) Sequence 320; May 17, 2010

Nozzle No.48; Sequence 320 TDPT; May 17, 2010

Nozzle No.4; Re-machine Sequence 200 (Pre-weld); May 14, 2010

Nozzle No.60; VT for Indication Interface Location; May 7, 2010

Nozzle No.33; VT Ind. @ 165° below J Groove; May 17, 2010

Nozzle No.60; VT-1; May 8, 2010

Nozzle No.24; Sequence 320-01; May 24, 2010

Nozzle No.28; Sequence 320-01; May 25, 2010

Nozzle No.10; Sequence 320-01 TDPT; May 25, 2010

Nozzle No.43; PT (Post-weld) Sequence 320; May 24, 2010

Nozzle No.58; PT (Post-weld) Sequence 320; May 25, 2010

Nozzle Nos.51, 55, 57, 58, 59, 60, 61, 64, 66, 67, VT Exam Sequence 360;
May 22, 2010

Nozzle No.4; PT (Post-weld) Sequence 320; May 25, 2010

Nozzle No.51; PT (Post-weld) Sequence 320; May 26, 2010

Nozzle No.55; PT (Post-weld) Sequence 320; May 27, 2010

Nozzle No.58; PT (Post-weld) Sequence 320-02; May 27, 2010

Nozzle No.61; PT (Post-weld) Sequence 320-01; May 27, 2010

Nozzle No.58; PT (Post-weld) Sequence D40-00; May 28, 2010

Nozzle No.4; VT (Post-weld) Sequence q. 320-01; May 28, 2010

Nozzle No.67; PT (Post-weld) Sequence 320-01; May 28, 2010

Nozzle No.58 PT repair Sequence D130-00; May 24, 2010

Nozzle No.58; VT (Post-weld) Sequence 360; May 28, 2010

NozzleNo.60; Sequence B10-00 TDPT; May 29, 2010

Nozzle No.64; PT (Post-weld) Sequence B10-00; May 29, 2010

Nozzle No.4; PT (Post-weld) Sequence B10-00; May 29, 2010

Nozzle No.59; PT (Post-weld) Sequence 320-01; May 29, 2010

Nozzle No.48; PT (Post-weld) Sequence B10-00; May 30, 2010

Nozzle No.57; PT (Post-weld) Sequence B10-00; May 29, 2010

Nozzle No.66; PT (Post-weld) Sequence B10-00; May 30, 2010

Nozzle No.40; PT (Post-weld) Sequence B10-00; May 30, 2010

Nozzle No.10; PT (Post-weld) Sequence B10-00; May 30, 2010

Nozzle No.29; PT (Post-weld) Sequence B10-00; May 30, 2010

Nozzle No.4; PT Repair Sequence C40-00; May 30, 2010

Nozzle No.21; PT (Post-weld) Sequence B10-00; May 30, 2010

Nozzle No.4; Final Repair Sequence C130-00; May 31, 2010

Nozzle No.27; PT (Post-weld) Sequence B10-00; May 31, 2010

Nozzle No.3; PT (Post-weld) Sequence B10-00; May 31, 2010

Nozzle No.62, PT (Post-weld) Sequence B10-00; May 31, 2010

Nozzle No.1; PT (Post-weld) Sequence B10-00; May 31, 2010

Nozzle No.16; PT (Post-weld) Sequence B10-00; June 1, 2010

Nozzle No.28; PT (Post-weld) Sequence B10-00; May 31, 2010

Nozzle No.43; Sequence B10-00 IDPT; May 31, 2010

Nozzle No.33; Sequence B10-00 IDPT; June 1, 2010

Nozzle No.1; VT Sequence 360; June 1, 2010

Nozzle No.48; VT Exam Sequence 360 QC; May 28, 2010

Nozzle No.4; VT Post Machine Bottom-Up; May 22, 2010

Nozzle Nos.1, 3, 10, 16, 21, 24, 33, 27, 28, 29, 40, 43; VT Exam Sequence 360; May 22, 2010

Welding Records

WPS; WP3/43/F43TBSCa3-005; Machine Temper Bead GTAW; Revision 5

PQR; PQ7183-03; Revision 3

PQR; PQ7295-000; dated May 21, 2010

PQR; PQ7293-001; dated October 16, 2009

PQR; PQ7296-000; dated June 15, 2010

55-PS0139-000; Procedure Supplement for PSO139-000; dated April 22, 2010

PS0139-002; Procedure Supplement for use in Conjunction with CRDM Penetration 4 Procedure Supplement PS0140 for WPS WP3/43/F43TBSCa3; dated May 13, 2010

PS0140-001; Procedure Supplement for use in Conjunction with CRDM Penetration 4 Procedure Supplement PS01390 for OI-0031; dated May 13, 2010

PS0140-002; Procedure Supplement for use in Conjunction with CRDM Penetration 4 Procedure Supplement PS01390 for OI-0031; dated May 21, 2010

PS0140-003; Procedure Supplement for use in Conjunction with CRDM Penetration 4 Procedure Supplement PS01390 for OI-0031; dated June 15, 2010

WPS; WP3/43/F43TBSCa3-005; Revision 5

50-9134306; Davis-Besse CRDM Nozzle IDTB Repair; Revision 2

AREVA WPQ; Welder A3265; dated November 10, 2009

AREVA WPQ; Welder C2648; dated February 9, 2005

AREVA WPQ; Welder L8350; dated March 19, 2008

AREVA WPQ; Welder O5503; dated February 17, 2007

AREVA WPQ; Welder S0524; dated April 19, 2010

AREVA NDE Certificate of Personnel Qualification; ID No: C5341; Revision 25

L-500466; Nozzle No.4, Alloy 82; dated May 24, 2010

L-500466; Nozzle No.4, Alloy 52M; dated May 23, 2010

CMTR 8416C: P.O. K256-0088, SA240 Type 304/304L; dated September 23, 2002

CMTR 07195702; Inconel Filler Metal 52M .035 x 10Spl; dated November 9, 2008

CMTR 07580702; Inconel Filler Metal 82 .035 x 10Spl; dated June 27, 2008

PQR; PQ7183-03; Revision 3

PQR; PQ7295-000 (for PS0140-002); dated May 21, 2010

PQR; PQ7293-001; dated October 16, 2009

<u>LIST OF ACRONYMS USED</u>

AR	Action Request
ASME	American Society for Mechanical Engineers
AWJR	Abrasive Water Jet Remediation
B&W	Babcock and Wilcox
BMV	Bare Metal Visual
CC	Code Case
CFR	Code of Federal Regulations
CRDM	Control Rod Drive Mechanism
EDY	Effective Degradation Years
EFPY	Effective Full Power Years
EPRI	Electric Power Research Institute
ET	Eddy Current
FZI	Fusion Zone Indication
^{o}F	Degrees Fahrenheit
gpm	Gallons per Minute
ICI	Incore Instrument
IMC	Inspection Manual Chapter
LER	Licensee Event Report
LOF	Lack of Fusion
uCi/cc	Micro-Curies per Cubic Centimeter
MRP	Material Reliability Program
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NRC	United States Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PDI	Performance Demonstration Initiative
PNNL	Pacific Northwest National Laboratory
PQR	Procedure Qualification Record
Psig	Pounds per Square Inch Gage
PT	Dye Penetrant
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Crack
QC	Quality Control
RCS	Reactor Coolant System
RCT	Root Cause Team
RIY	Reinspection Year
RFO	Refueling Outage
RR	Relief Request
RVCH	Reactor Vessel Closure Head
SDP	Significance Determination Process
SEM	Scanning Electron Microscope
SIT	Special Inspection Team
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Examination

CHRONOLOGY OF EXAMINATIONS

Table 1 – Reactor Vessel Closure Head Examination History

Examination Dates	Type of Exam	NRC Requirement	Results/Comments
8/4/1975 - 8/6/1975	Preservice Hydrostatic Pressure Test	American Society of Mechanical Engineers Code (ASME) Section III	Pressure test at 3125 psig. No Leakage Identified
7/15/1975	Preservice dye penetrant (PT) examination of control rod drive mechanism (CRDM) nozzle J-groove welds	ASME Code Section III	Rounded indications identified at nozzles 28, 29, and 62, and linear indications for nozzles 46, 51, 58, 63, and 64. Defects were removed by probe grinding, and repaired before final PT. The final PT examination results were acceptable
6/10/2002 – 6/12/2002	PT of CRDM nozzle J-groove welds	None. Licensee initiative, exam performed to supplement ASME Code Data Packages	PT examination results were acceptable. Small rounded indications identified on 7 nozzles, which met the procedure and ASME Code acceptance requirements
7/3/2002 – 7/4/2002	PT of CRDM nozzle J-groove welds	None. Licensee initiative, exam performed to supplement ASME Code Data Packages	Examinations conducted to record/photograph relevant indications. PT examination results were acceptable. Small rounded indications on 8 nozzles, which met the procedure and ASME Code acceptance requirements
6/22/2002 – 7/7/2002	Ultrasonic (UT) examination of CRDM nozzles	None. Licensee initiative, exam performed to supplement ASME Code Data Packages	66 nozzles had weld fabrication indications and 8 nozzles had recordable nozzle base-metal indications. No specific acceptance criteria applied to these fabrication indications

CHRONOLOGY OF EXAMINATIONS

Examination Dates	Type of Exam	NRC Requirement	Results/Comments
6/4/2002 – 7/3/2002	Baseline eddy current examination of the inside surface of CRDM nozzles. This exam did not include the J-groove welds	None. Licensee initiative	Eddy current examination performed to meet ASME Code, Section XI, 1995 Edition, 1996 Addenda, and industry standards. No rejectable indications. Recordable indications identified in 5 nozzles
7/12/2002	Visual inspection (VT-1) of the vessel head after cleaning	None. Licensee initiative	Results acceptable. Minor pitting noted on 13 CRDM nozzles
10/14/2003 – 10/17/2003	Bare metal visual examination of the reactor vessel head	None. Licensee initiative	No evidence of leakage. Light scattered debris was easily removed with air. Licensee stated component cooling water leakage from above the head may have caused all the indications found
1/22/2005 – 1/23/2005	Mid-Cycle Outage - Bare metal visual examination of the reactor vessel head	NRC Confirmatory Order EA-03-214	No leakage identified. A small amount of staining was present that existed before the current operating cycle
3/14/2006 – 3/16/2006	Refueling Outage 14 - Bare metal visual examination of the reactor vessel head	Commitments to the NRC: 1. Davis-Besse response to NRC Bulletin 2003-02 (ML033250499) 2. Davis-Besse Consent to Revised Order EA-03-009 (ML041610291) 3. February 3, 2005, Letter to NRC concerning Mid-Cycle 14 Inspection Results (ML050350194)	No evidence of leakage

CHRONOLOGY OF EXAMINATIONS

Examination Dates	Type of Exam	NRC Requirement	Results/Comments
1/7/2008 – 1/8/2008	Refueling Outage 15 - Bare metal visual examination of the reactor vessel head	NRC Order EA-03-009 Reference- Davis-Besse Consent to Revised Order EA-03-009 (ML041610291)	No evidence of leakage
3/12/2010 – 3/30/2010	Refueling Outage 16 - Bare metal visual examination of the reactor vessel head UT examination of CRDM nozzles PT examinations of 11 CRDM nozzle locations	10 CFR 50.55a(g)(6)(ii)(D) ASME Code Case N-729-1	12 nozzles with recordable indications based on UT examinations 1 nozzle with active leakage based on bare metal visual examination. No wastage detected on vessel head 13 nozzles with potential leakage based on bare metal visual examination 4 J-groove welds with rejectable indications based on PT examinations
4/28/2010 – 5/1/2010	Eddy current examinations of the J-groove weld surface for 51 CRDM nozzle locations	None Licensee initiative	8 nozzles were identified with rejectable indications

<u>COMPARISON OF EXAMINATION RESULTS</u>

Table 1- Bare Metal Visual (BMV) Examination Comparison

Nozzle No. or Exam Area and Team Conclusions	BMV Exam Results - 2010 As-found (Note 1)	BMV Exam Results – 2008 (Note 2)	BMV Exam Results – 2006 (Note 2)	BMV Exam Results – 2005 (Note 2)	BMV Exam Results – 2003 (Note 2)	Preservice Hydrostatic Test – 1975 (Note 2)
No. 4 No pre-existing conditions.	Active leakage – 360 degrees around nozzle penetration. Heavy buildup of deposits and borated water stains with a glazed coating on nozzle penetration and base material	No evidence of leakage	No evidence of leakage	No evidence of leakage	No evidence of leakage	Acceptable results
No. 11 No pre-existing conditions.	Potential active leakage. Boric acid deposits in 2 quadrants. C quadrant had tightly adherent white deposits	No evidence of leakage	No evidence of leakage	No evidence of leakage	No evidence of leakage	Acceptable results
No. 12 No pre-existing conditions.	Potential active leakage. Coating of white borated deposits 360 degrees around nozzle	No evidence of leakage	No evidence of leakage	No evidence of leakage	No evidence of leakage	Acceptable results

COMPARISON OF EXAMINATION RESULTS

Nozzle No. or Exam Area and Team Conclusions	BMV Exam Results - 2010 As-found (Note 1)	BMV Exam Results – 2008 (Note 2)	BMV Exam Results – 2006 (Note 2)	BMV Exam Results – 2005 (Note 2)	BMV Exam Results – 2003 (Note 2)	Preservice Hydrostatic Test – 1975 (Note 2)
No. 16 No pre-existing conditions.	Potential active leakage. Several adherent white deposits around nozzle	No evidence of leakage	No evidence of leakage	No evidence of leakage	No evidence of leakage	Acceptable results
No. 28 No pre-existing conditions.	Potential active leakage. Tightly adherent coating of white deposits noted in annulus region 360 degrees around nozzle	No evidence of leakage	No evidence of leakage	No evidence of leakage	No evidence of leakage	Acceptable results
No. 33 No pre-existing conditions.	Potential active leakage. Boric acid deposits in quadrants A and B. Quadrants C and D have white stains on base material	No evidence of leakage	No evidence of leakage	No evidence of leakage	No evidence of leakage	Acceptable results

COMPARISON OF EXAMINATION RESULTS

Nozzle No. or Exam Area and Team Conclusions	BMV Exam Results - 2010 As-found (Note 1)	BMV Exam Results – 2008 (Note 2)	BMV Exam Results – 2006 (Note 2)	BMV Exam Results – 2005 (Note 2)	BMV Exam Results – 2003 (Note 2)	Preservice Hydrostatic Test – 1975 (Note 2)
No. 40 No pre-existing conditions.	Potential active leakage Tightly adherent white deposits noted on quadrant B	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	Acceptable results
No. 48 No pre-existing conditions.	Potential active leakage Adherent white deposits in annulus on quadrant B. Quadrants A, C and D have white stains	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	Acceptable results
No. 51 No pre-existing conditions.	Potential active leakage White residue in annulus at quadrant D	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	Acceptable results

COMPARISON OF EXAMINATION RESULTS

Nozzle No. or Exam Area and Team Conclusions	BMV Exam Results - 2010 As-found (Note 1)	BMV Exam Results – 2008 (Note 2)	BMV Exam Results – 2006 (Note 2)	BMV Exam Results – 2005 (Note 2)	BMV Exam Results – 2003 (Note 2)	Preservice Hydrostatic Test – 1975 (Note 2)
No. 53 No pre-existing conditions.	Potential active leakage. Adherent white deposits in annulus and on base material in quadrants A and B.	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	Acceptable results.
No. 57 No pre-existing conditions.	Potential active leakage. Adherent white deposits in annulus on quadrants A, C and D.	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	Acceptable results.
No. 58 No pre-existing conditions.	Potential active leakage. White residue on base material and in annulus area on quadrant D.	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	Acceptable results.
No. 60 No pre-existing conditions.	Potential active leakage. Adherent white deposits in annulus on quadrant B with rust staining.	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	Acceptable results.

COMPARISON OF EXAMINATION RESULTS

Nozzle No. or Exam Area and Team Conclusions	BMV Exam Results - 2010 As-found (Note 1)	BMV Exam Results – 2008 (Note 2)	BMV Exam Results – 2006 (Note 2)	BMV Exam Results – 2005 (Note 2)	BMV Exam Results – 2003 (Note 2)	Preservice Hydrostatic Test – 1975 (Note 2)
No. 64 No pre-existing conditions.	Potential active leakage. Adherent white deposits in annulus on quadrants A, B, and D.	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	No evidence of leakage.	Acceptable results.
Head surface area as described in ASME Code Case N-729-1 No pre-existing conditions.	Boric acid type deposits indicative of leakage were identified at multiple nozzles and on multiple areas of the vessel head surface.	Isolated deposits confirmed not to be boric acid. Rust stains attributed to a component cooling water leak. Several nozzles had as-found boric acid deposits. After application of air, these deposits were cleared.	No evidence of boric acid leakage or deposits.	Small build-up of material and stains near nozzles and at some head-to-nozzle interfaces. Material is orange-brown in color with very little volume. Material and stains existed before the current operating cycle. No material wastage identified.	Lightly scattered debris on about 80 percent of the head surface. Debris was easily removed with 60 psi air application. No leakage or material wastage was identified.	Acceptable results.

Notes for Table 1 - Bare Metal Visual (BMV) Examination Comparison

Note 1 Based on the Special Inspection Team's direct observation of the BMV examination and review of examination photographs, video recordings, and written records.

Note 2 Based on the Special Inspection Team's review of written examination records as supplemented by examination photographs and video records.

COMPARISON OF EXAMINATION RESULTS

Table 2 - J-Groove Weld Dye Penetrant (PT) or Eddy Current (ET) Examination Comparison

J-Groove Weld Nozzle Location	2010 Examination Type and Results	2002 Baseline – PT prior to service	1975 Pre-service PT (fabrication records)	Team Conclusions/Comments
No. 1	ET examination identified a rejectable indication. One indication – rounded 0.160 inch diameter.	PT acceptable.	PT acceptable.	Not a pre-existing flaw. However, different methods (ET verses PT) may have different sensitivity to this type of flaw. Flaw met licensee acceptance criteria, but licensee elected to repair this nozzle.
No. 3	ET examination identified a non- quantifiable indication. One indication located at the J-groove weld to clad interface which extended from 59 to 116 degrees.	PT acceptable.	PT acceptable.	Not likely a pre-existing flaw. However, different methods (ET verses PT) may have different sensitivity to this type of flaw. The indication at the weld to cladding interface was confirmed by a visual examination and the licensee elected to repair this nozzle.
No. 16	PT examination identified a rejectable indication (No. 2). Indication 1 - rounded, 0.10 inch diameter with light bleedout. Indication 2 - rounded, 0.15 inch diameter moderate bleedout.	PT acceptable.	PT acceptable.	Were not pre-existing flaws.

COMPARISON OF EXAMINATION RESULTS

J-Groove Weld Nozzle Location	2010 Examination Type and Results	2002 Baseline – PT prior to service	1975 Pre-service PT (fabrication records)	Team Conclusions/Comments
No. 21	ET examination identified a rejectable indication. Indication 1 – circumferential, 0.130 inch length.	PT acceptable.	PT acceptable.	Not likely a pre-existing flaw. However, different methods (ET verses PT) may have different sensitivity to this type of flaw.
No. 27	ET examination identified a rejectable indication. Indication 1 – linear 0.180 inch in length.	PT acceptable.	PT acceptable.	Not likely a pre-existing flaw. However, different methods (ET verses PT) may have different sensitivity to this type of flaw.
No. 29	ET examination identified one rejectable indication. Indication 1 – linear 0.120 inch in length.	PT acceptable.	PT acceptable. Rounded indication during initial PT. The defect was removed and final PT was acceptable.	Not likely a pre-existing flaw. However, different methods (ET verses PT) may have different sensitivity to this type of flaw. Also, the 1975 pre-service indication is in a different location as the 2010 ET indication. Furthermore, the pre-service indication was removed prior to the final PT in 1975. Subsequent PTs in 1975, 2002 and 2010 did not reveal the indication.

COMPARISON OF EXAMINATION RESULTS

J-Groove Weld Nozzle Location	2010 Examination Type and Results	2002 Baseline – PT prior to service	1975 Pre-service PT (fabrication records)	Team Conclusions/Comments
No. 40	PT examination identified two recordable indications. Indication 1- rounded, 0.125 inch diameter located in cladding 1 inch from nozzle face. Indication 2 - rounded, 0.125 inch diameter, located in cladding, 1 inch from nozzle face. ET examination confirmed two rejectable indications at the same locations as the PT examination. Indication 1- circumferential, 0.085 inch in length, correlating with PT indication 1. Indication 2 – circumferential, 0.105 inch in length, correlating with PT indication 2.	PT acceptable, with 1 rounded indication with a size of 0.05 inch located 0.4 inch from nozzle face.	PT acceptable.	Were not likely pre-existing flaws. The 2002 PT indication was located in the same general area as the indication 2 in 2010. However, the 2010 indication was measured to be 1 inch from nozzle face and the baseline indication was measured at 0.4 inch from the nozzle face. Licensee elected to reject these indications and repair this nozzle.

COMPARISON OF EXAMINATION RESULTS

J-Groove Weld Nozzle Location	2010 Examination Type and Results	2002 Baseline – PT prior to service	1975 Pre-service PT (fabrication records)	Team Conclusions/Comments
No. 48	PT examination identified rejectable indications. Indication 1- linear, 0.15 inch in length. Indication 2 - linear, 0.35 inch in length. Indication 3 - linear, 0.20 inch in length.	PT acceptable.	PT acceptable.	Were not pre-existing flaws.
No. 53	PT examination identified two recordable indications. Indication 1 - rounded, 0.125 inch diameter. Indication 2 - rounded, 0.10 inch diameter. Indications removed by grinding. Subsequent PT examinations confirmed indications were removed.	PT acceptable.	PT acceptable.	Were not pre-existing flaws. Licensee accepted this nozzle for continued service following removal of the PT indications by grinding.

COMPARISON OF EXAMINATION RESULTS

J-Groove Weld Nozzle Location	2010 Examination Type and Results	2002 Baseline – PT prior to service	1975 Pre-service PT (fabrication records)	Team Conclusions/Comments
No. 57	PT examination identified rejectable linear and rounded indications (Nos. 1 and 2). Indication 1 -rounded, 0.20 inch diameter. Indication 2 -linear, 0.15 inch length. Indication 3 -rounded, 0.06 inch diameter	PT acceptable.	PT acceptable.	Were not pre-existing flaws.
No. 60	PT examination identified a recordable indication. Indication 1 - rounded, 0.125 inch diameter. ET examination indications identified. Indication1 - circumferential, 0.090 inch in length. Indication 2 – non-relevant indication that correlates with PT indication 1.	PT acceptable.	PT acceptable.	Not a pre-existing flaw. Licensee elected to reject the PT flaw indication based on small circumferential ET indication identified.
J-Groove Weld Nozzle Location	2010 Examination Type and Results	2002 Baseline – PT prior to service	1975 Pre-service PT (fabrication records)	Team Conclusions/Comments

COMPARISON OF EXAMINATION RESULTS

No. 64	PT examination identified a rejectable linear flaw (No. 2). Indication 1 - rounded, 0.0625 inch Indication 2 - linear, 0.375 inch.	PT acceptable.	PT acceptable. Linear indication during initial PT. The defect was removed and final PT was acceptable.	Potentially a pre-existing flaw that propagated from fabrication repair area during service. During the 1975 fabrication work on an indication was identified and was repaired (repair cavity prior to fabrication weld repairs was 2.5 inches in length, 1.125 inches wide and 0.125 inch deep). The 1975 pre-service indication and repair work was in the same location on the J-groove weld as the 2010 indication. However, after this indication was removed, subsequent PTs in 1975 and 2002 did not reveal the indication. Therefore, it was possible this 2010 flaw indication had propagated inservice from the 1975 fabrication repair area.
No. 66	PT examination identified a recordable indication. Indication1 - rounded, 0.125 inch diameter. ET examination identified a rejectable indication. Indication1 – axial, 0.200 inch in length that correlated with PT indication 1.	PT acceptable.	PT acceptable.	Not a pre-existing flaw.

COMPARISON OF EXAMINATION RESULTS

Table 3 - Ultrasonic (UT) Examination Comparison

Nozzle Location	2010 recorded UT data (Note 1)	Conclusions on evidence of pre-existing flaw based on a review of the 2002 preservice UT data (Note 2)
No. 4	Three axially oriented outside diameter (OD) surface UT PWSCC indications identified at or near J-groove weld toe. Indication No. 2 traverses height of J-groove weld. A 1 - L= 1.450, D = 0.380 A 2 - L= 1.465, D = 0.408 C 2 - L = Not Recorded, D=0.384 (this axial PWSCC indication has a small circumferential component) A 3 - L= 0.758, D = 0.250 Leakage indication on backwall sensitive channel.	Indication No. 2 could possibly have possibly propagated from an original fabrication weld indication. Specifically, the licensee report documented a low amplitude circumferential reflector in baseline UT data that suggests an original reflector (lack of fusion, slag, repair weld) as the origin for the circumferential component of indication No. 2. The licensee concluded that the increase in signal response for this reflector could be the result of propagation from indication No 2. The team did not identify flaw like indications in the preservice UT data at flaw locations identified in the 2010 UT data. No evidence of leakage path in pre-service records based on review of zero degree search unit data.
No. 10	Four axially oriented OD surface UT PWSCC indications identified at or near toe of J-groove weld elevation. A 1 - L= 0.260, D = 0.184 A 2 - L= 0.223, D = 0.149 A 3 - L= 0.817, D = 0.307 A 4 - L= 0.446, D = 0.183	No flaw like indications identified in preservice UT data at flaw locations identified in 2010 UT data. No evidence of leakage path in pre-service records based on review of zero degree search unit data.

COMPARISON OF EXAMINATION RESULTS

Nozzle Location	2010 recorded UT data (Note 1)	Conclusions on evidence of pre-existing flaw based on a review of the 2002 preservice UT data (Note 2)
No. 24	One axial oriented OD surface UT PWSCC indication identified at or near toe of the J-groove weld elevation. A 1- L= 0.702, D = 0.209	No flaw like indications identified in preservice UT data at flaw location identified in 2010 UT data. No evidence of leakage path in pre-service records based on review of zero degree search unit data.
No. 28	Two axially oriented OD surface UT PWSCC indications. Indication No. 1 traverses height of J-groove weld. Indication No 2 at or near toe of J-groove weld elevation. A 1- L= 1.933, D = 0.612 A 2 - L= 1.117, D = 0.367 Inside surface anomalies also recorded (e.g. shallow indications which may be indicative of inside surface scratches).	No flaw like indications identified in preservice UT data at flaw locations identified in 2010 UT data. No evidence of leakage path in pre-service records based on review of zero degree search unit data.
No. 33	Single axially oriented OD surface UT PWSCC indication traversing height of J-groove weld. A 1 L= 2.008, D = 0.632	No flaw like indications identified in preservice UT data at flaw location identified in 2010 UT data. No evidence of leakage path in pre-service records based on review of zero degree search unit data.

COMPARISON OF EXAMINATION RESULTS

Nozzle Location	2010 recorded UT data (Note 1)	Conclusions on evidence of pre-existing flaw based on a review of the 2002 preservice UT data (Note 2)
No. 43	Single axial OD surface UT PWSCC indication at or near J-groove weld toe elevation. A 1 L= 0.409, D = 0.190 Inside surface anomalies also recorded (e.g., shallow indications which may be indicative of inside surface scratches).	No flaw like indications identified in preservice UT data at flaw location identified in 2010 UT data. No evidence of leakage path in pre-service records based on review of zero degree search unit data.
No. 51	Single circumferential OD surface UT PWSCC indication at or near J-groove weld toe elevation. C 1 L= 0.427, D = 0.198	No flaw like indications identified in preservice UT data at flaw location identified in 2010 UT data. The team noted that a Code acceptable PT indication (from 2002 baseline PT record) was identified at the toe of J-groove weld within a 30 degrees azimuth of the UT indication location. No evidence of leakage path in pre-service records based on review of zero degree search unit data.
No. 55	Two axially oriented OD surface UT PWSCC indications identified. Indication No. 1 traverses height of J-groove weld. Indication No. 2 is at or near the J-groove weld toe elevation. Indications are separated by less than 10 degrees in nozzle circumference. A 1 L= 1.686, D = 0.521 A 2 L= 0.667, D = 0.521 Inside surface anomalies also recorded (e.g. shallow indications which may be indicative of inside surface scratches.) VT-1 identified scratches in this area.	No flaw like indications identified in preservice UT data at flaw locations identified in 2010 UT data. No evidence of leakage path in pre-service records based on review of zero degree search unit data.

COMPARISON OF EXAMINATION RESULTS

Nozzle Location	2010 recorded UT data (Note 1)	Conclusions on evidence of pre-existing flaw based on a review of the 2002 preservice UT data (Note 2)
No. 58	Single axial OD surface UT PWSCC indication at or near J-groove weld toe elevation. A 1 L= 0.451, D = 0.161	No flaw like indications identified in preservice UT data at flaw location identified in 2010 UT data.
No. 59	Single axial OD surface UT PWSCC indication that traverses height of J-groove weld. A 1 L= 1.796, D = 0.549	No flaw like indications identified in preservice UT data at flaw location identified in 2010 UT data. No evidence of leakage path in pre-service records based on review of zero degree search unit data.
No. 61	Single axial OD surface UT PWSCC indication at or near J-groove weld toe elevation. A 1 L= 1.043, D = 0.346 Inside surface anomalies also recorded (e.g. shallow indications which may be indicative of inside surface scratches.) VT-1 identified scratches in this area.	No flaw like indications identified in preservice UT data at flaw location identified in 2010 UT data. No evidence of leakage path in pre-service records based on review of zero degree search unit data.
No. 67	Single axial OD surface UT PWSCC indication at J-groove weld. Indication is located near center of J-groove weld elevation. A 1 L= 0.297, D = 0.156 Leakage indication on backwall sensitive channel, but not a typical of leakage pattern.	No flaw like indications identified in preservice UT data at flaw location identified in 2010 UT data. No evidence of leakage path in pre-service records based on review of zero degree search unit data.

COMPARISON OF EXAMINATION RESULTS

Notes for Table 3 - Ultrasonic Examination Data Comparison

Note 1 A= Axially oriented indication direction -aligned with tube axis, but indications may be several degrees off this axis and still be classified as axial.

C = Circumferentially oriented indication - aligned with tube circumference, but indications may be several degrees off this axis and still be classified as circumferential.

L = Length of indication (inches) using search unit sensitive to primary flaw direction (e.g. axial sensitive probe for axial flaws or circumferential sensitive probe for circumferential flaws)

D = Depth of indication (inches) referenced from the outside diameter (OD) of the CRDM nozzle tube wall (0.63 inch thick) using search unit sensitive to primary flaw direction.

Note 2 The team compared preservice UT data to UT data acquired in the 2010 refueling outage. Comparison was based on zero degree search unit (for leak path) and transducers with similar angles for crack indications.

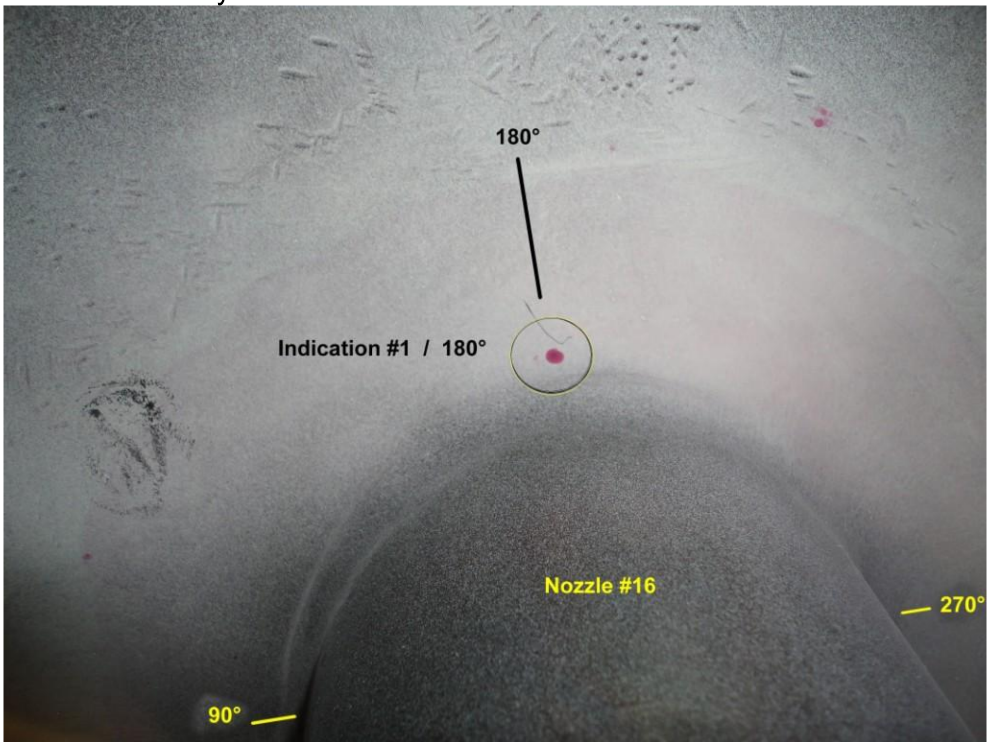
PICTURES - EXAMINATIONS, NOZZLE REPAIRS, AND FLOW PATHS

Picture No. 1A - As-Found Bare Metal Visual – Nozzle 4 (A Quadrant)

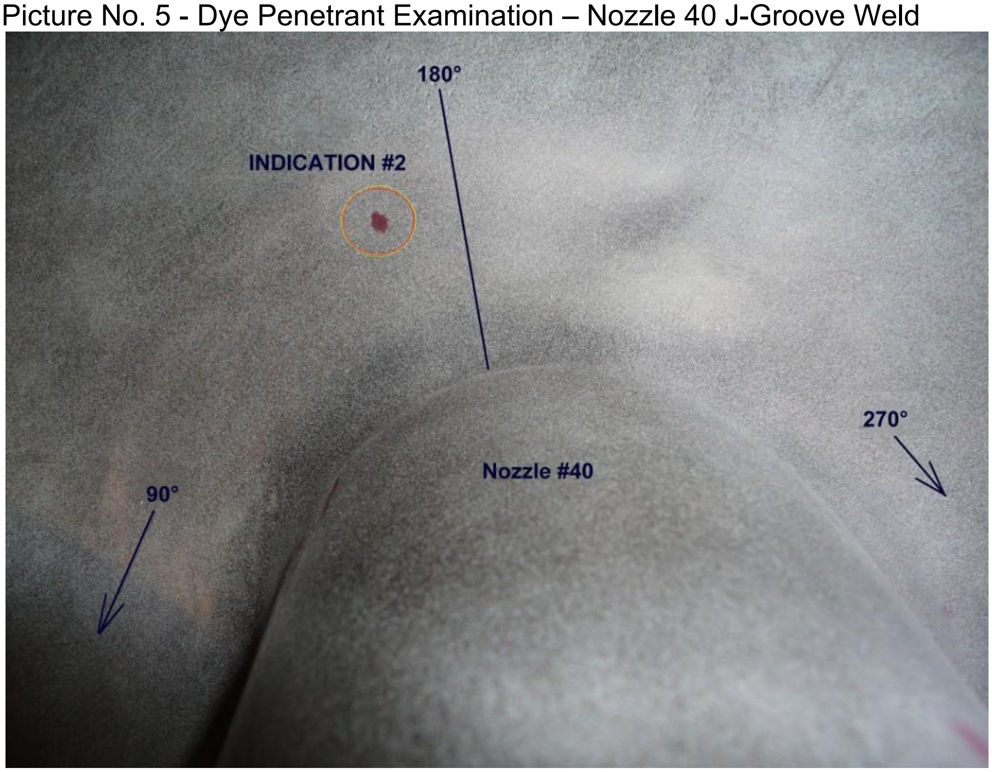
Picture No. 2 – As-Found Bare Metal Visual – Nozzle 12 (C Quadrant)

Picture No. 3 - As-Found Bare Metal Visual – Nozzle 33 (D Quadrant)

Picture No. 4 - Dye Penetrant Examination – Nozzle 16 J-Groove Weld

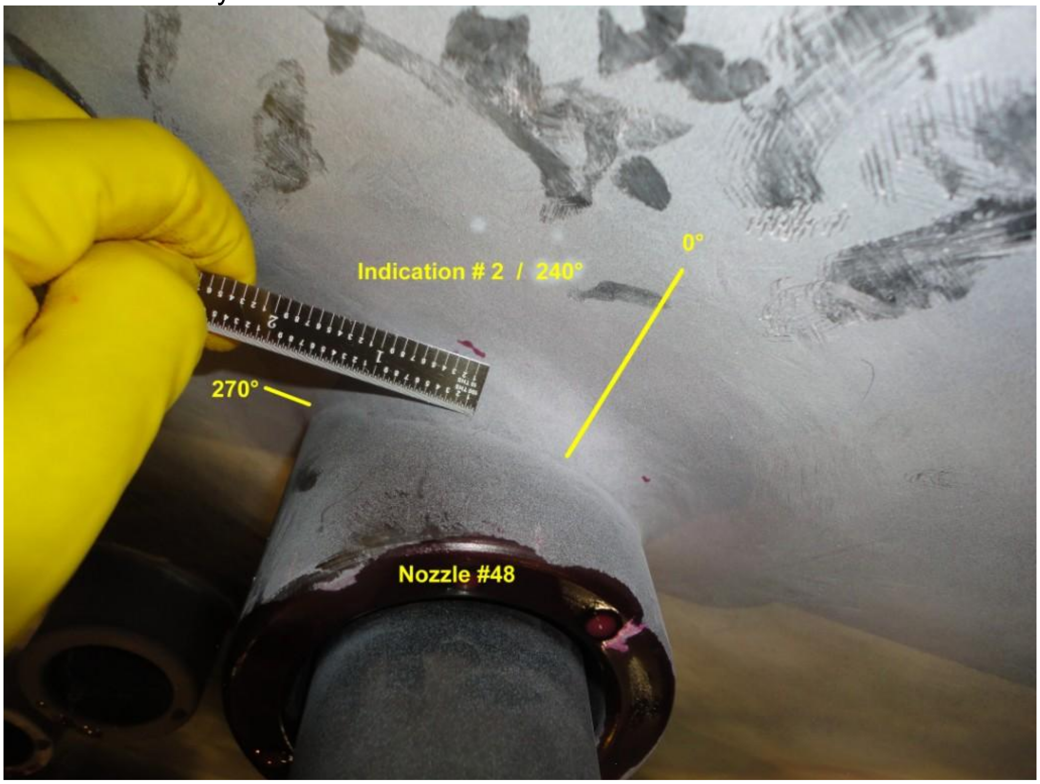


Picture No. 5 - Dye Penetrant Examination – Nozzle 40 J-Groove Weld

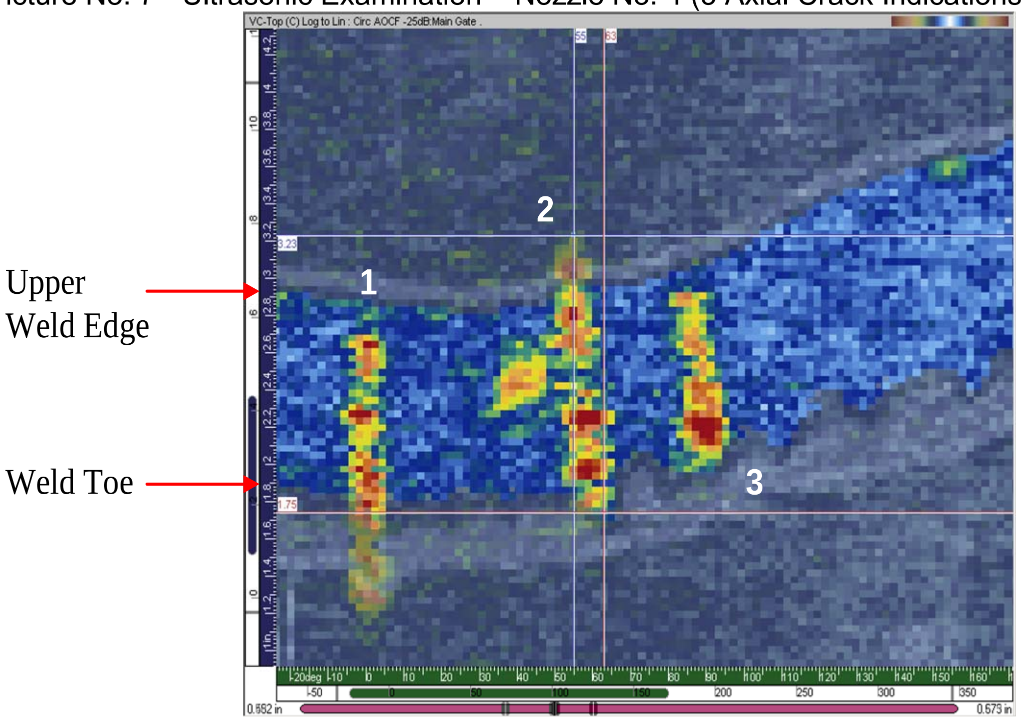


PICTURES - EXAMINATIONS, NOZZLE REPAIRS, AND FLOW PATHS

Picture No. 6 - Dye Penetrant Examination – Nozzle 48 J-Groove Weld

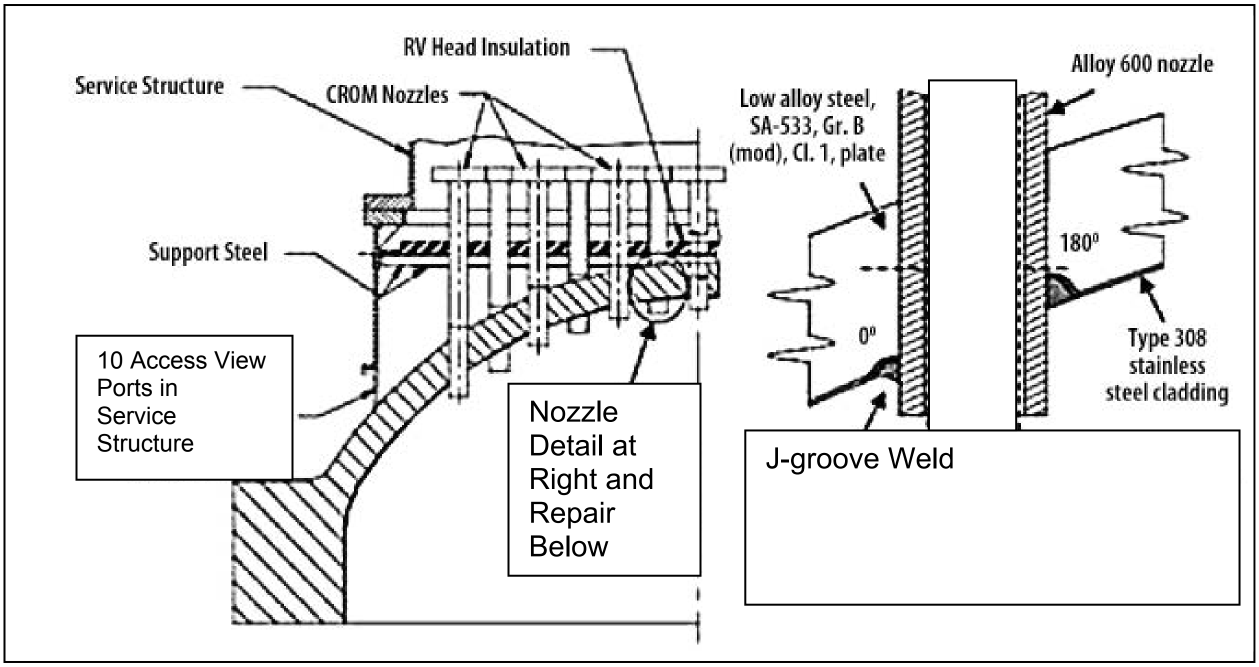


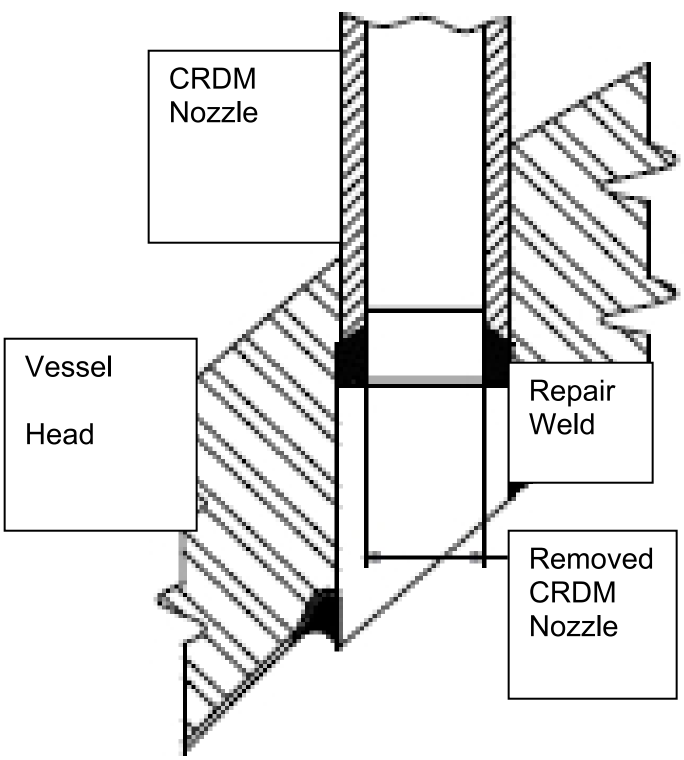
Picture No. 7 - Ultrasonic Examination – Nozzle No. 4 (3 Axial Crack Indications)



PICTURES - EXAMINATIONS, NOZZLE REPAIRS, AND FLOW PATHS

Picture No. 8 – Half-Nozzle Repair Diagram – Vertical Section Views



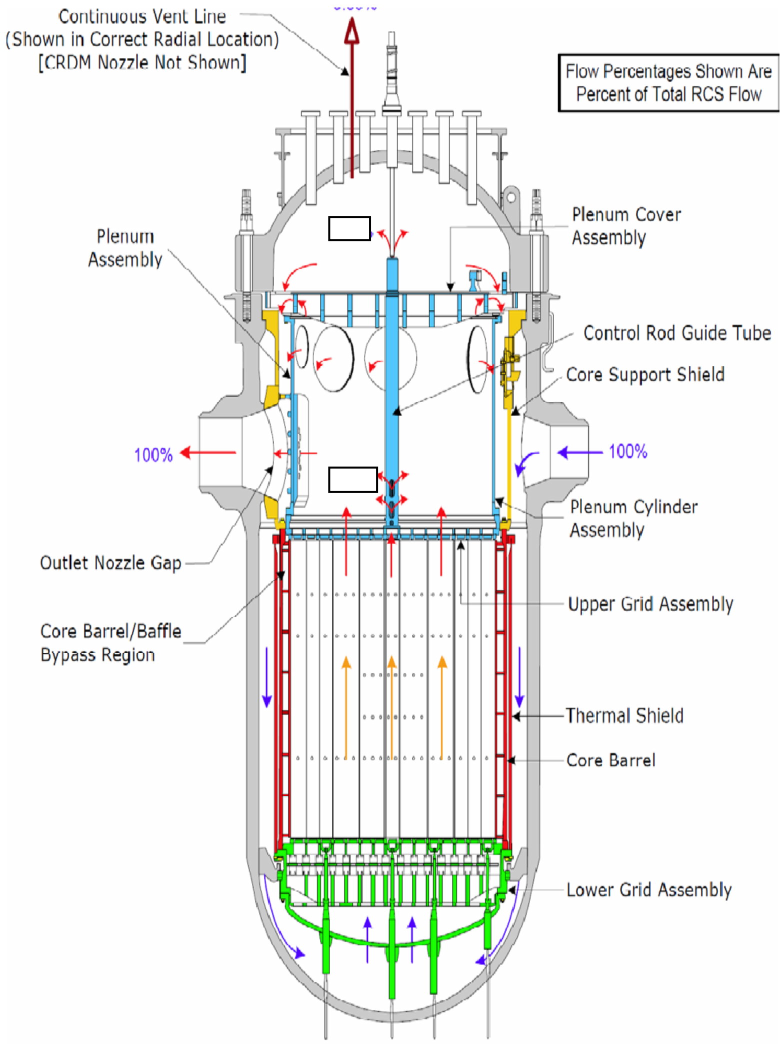


Picture No. 9 – Half-Nozzle Repair Mockup

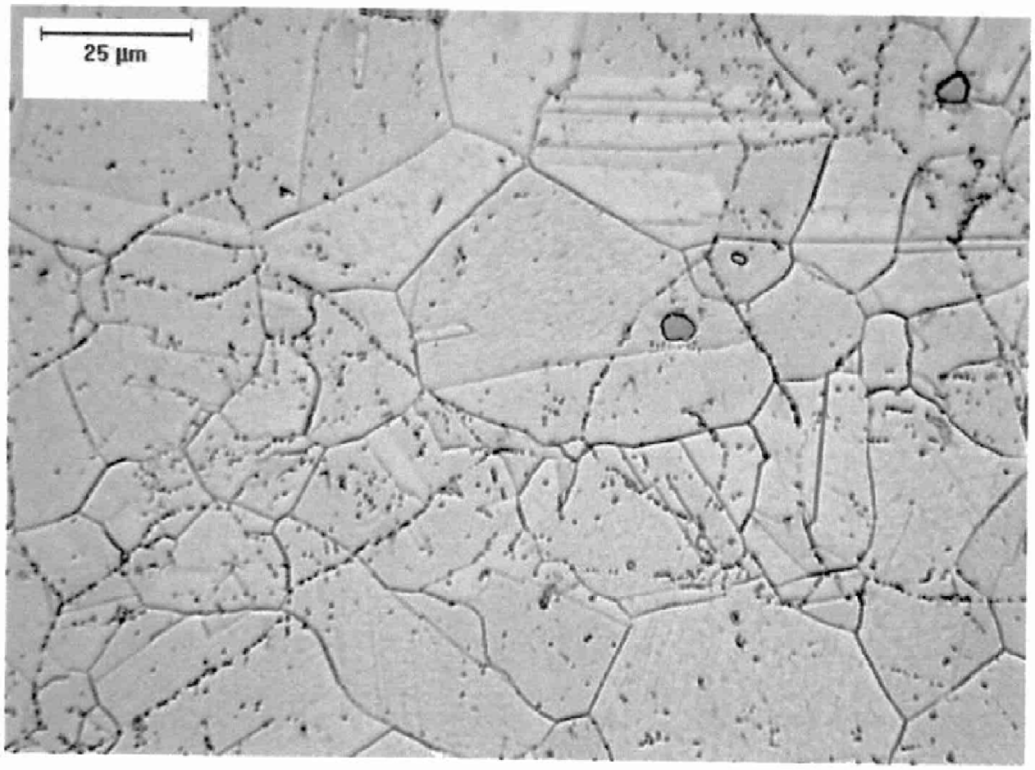
**Control Rod Drive Mechanism
Penetration Nozzle
Before Repair**

**Control Rod Drive Mechanism
Penetration Nozzle
After Repair**

Picture No. 10 – Vessel Flow Paths



Picture No. 11 – Random Carbide Distribution - Material Heat No. M7929



DAVIS-BESSE SPECIAL INSPECTION CHARTER

March 17, 2010

MEMORANDUM TO: Mel Holmberg, Senior Reactor Engineer
Division of Reactor Safety

FROM: Anne T. Boland, Director */RA/*
Division of Reactor Safety

SUBJECT: SPECIAL INSPECTION CHARTER TO REVIEW FLAWS IN THE CONTROL ROD DRIVE MECHANISM (CRDM) REACTOR VESSEL CLOSURE HEAD NOZZLE PENETRATIONS AT THE DAVIS-BESSE NUCLEAR POWER STATION

Beginning on March 12, 2010, during ultrasonic examinations, the licensee-identified indications of flaws in several CRDM nozzles, which penetrate the reactor vessel closure head (RVCH). In addition, the licensee-identified small amounts of boric acid residue around some nozzles, indicative of through wall leakage. The RVCH, procured as a replacement head from the uncompleted Midland Nuclear Power Plant, had been installed during the 2002/2003 extended shutdown, and placed into service in early 2004. The licensee had conducted bare metal visual examinations during a mid-cycle outage in 2005 and two refueling outages in 2006 and 2008 and had not identified any boric acid leakage. The March 2010 examinations were the first ultrasonic examinations performed on the replacement RVCH since it was placed in service.

Management Directive 8.3 and Inspection Manual Chapter 0309 Review

The facts and circumstances currently known about the CRDM flaws were evaluated against the criteria in Management Directive 8.3 and Inspection Manual Chapter (IMC) 0309. None of the deterministic criteria were specifically met. However, in accordance with IMC 0309, “Reactive Inspection Decision Basis for Reactors,” Section 04.04, “factors such as openness, public interest, and public safety should be appropriately considered by NRC when deciding whether to dispatch an IIT, AIT, or SI.” Given the history of Davis-Besse Nuclear Power Station with respect to the RVCH, the expected significant public interest, and the strong desire to ensure openness, the NRC has decided it is appropriate to conduct a Special Inspection of the CRDM nozzle flaws. The team will also communicate to the Office of Nuclear Reactor Regulation (NRR) staff any information as may be developed (e.g., the licensee’s root cause) and pertinent to evaluating possible generic industry implications and programmatic initiatives. The deterministic criteria will be re-evaluated as more information becomes available.

Special Inspection Activities

The Special Inspection will commence on March 16, 2010. The team will consist of Mel Holmberg, Senior Reactor Engineer (ISI specialist), Jay Collins, NRR Senior Materials Engineer, and Adam Wilson, Resident Inspector. The team will also consult with NRR technical staff, as well as Region III Senior Reactor Analysts as needed.

The primary consideration for the team is to independently assess and confirm the adequacy of the licensee's identification, analyses, and resolution of identified CRDM nozzle flaws to ensure the acceptability of placing the RVCH back in service. A charter was developed and is enclosed. An entrance meeting was conducted on Tuesday, March 16, 2010.

DAVIS-BESSE SPECIAL INSPECTION TEAM CHARTER

This Special Inspection is chartered to assess the circumstances surrounding the identification of flaws in the CRDM RVCH nozzle penetrations at Davis-Besse Nuclear Power Station. This Special Inspection will be conducted in accordance with Inspection Procedure 93812, "Special Inspection," and will include, but not limited to, the following items:

1. Establish the pertinent examination chronology/history of the replacement RVCH.

2. Compare current examination results with samples of the 2005 – 2008 examination records and pre-service records to determine whether the conditions were pre-existing.

3. Evaluate the adequacy of the licensee's plan for assessing the causes of the flaws and the licensee's rationale regarding acceptability of the head for continued service.

4. Review current examination results and monitor in-progress examination and analysis activities to ensure they are adequately conducted. Confirm based on review of the examination results, that the licensee has identified appropriate nozzles for repair and the acceptability of remaining nozzles for service.*

5. Evaluate the adequacy of the repair activities and monitor implementation. Confirm that the repair implemented complies with NRC requirements.*

*Given the overlap between this Special Inspection and baseline Inspection Procedure (IP) 71111.08, "Inservice Inspection Activities," Section 02.02, "PWR Vessel Upper Head Penetration Inspection Activities," ensure that inspection activities cover the scope of that Section and document/credit completion of that Section in the inspection report.

Based on the results of this inspection, three NRC-identified findings of very low safety significance were identified. Each finding involved a violation of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your Corrective Action Program, the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy (November 28, 2008). Additionally, during the previous operating cycle, the Davis-Besse Nuclear Power Station was operated with through-wall pressure boundary leakage from cracked control rod drive mechanism (CRDM) nozzles, which was contrary to the Technical Specification 3.4.13 requirement. Operation with pressure boundary leakage of this magnitude (below detection thresholds) would normally be considered a Severity Level IV violation. However, the staff has reviewed your root cause analysis of the event and has concluded that this equipment failure could not have been reasonably avoided or detected by your quality assurance program or other related control measures. Therefore, after consultation with the Regional Administrator, Region III, and the Director, Office of Enforcement, I have been authorized in accordance with Section VII.B.6 of the Enforcement Policy (November 28, 2008), to exercise enforcement discretion and not issue a violation for this issue.

If you contest the subject or severity of the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Davis-Besse Nuclear Power Station. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at Davis-Besse Nuclear Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

Ann Boland, Director
Division of Reactor Safety

Docket Nos. 50-346
License Nos. NPF-3

Enclosure: Inspection Report 05000346/2010-008(DRS);
w/Attachments:
1. Supplemental Information
2. Chronology of Examinations
3. Comparison of Examination Results
4. Pictures- Examinations, Nozzle Repairs, and Flow Paths
5. Davis-Besse Special Inspection Charter

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OFFICIAL RECORD COPY

Letter to Mr. Barry Allen from Ms. Anne T. Boland dated October 22, 2010.

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION SPECIAL INSPECTION TO REVIEW FLAWS IN THE CONTROL ROD DRIVE MECHANISM REACTOR VESSEL CLOSURE HEAD NOZZLE PENETRATIONS 05000346/2010-008(DRS) AND EXERCISE OF ENFORCEMENT DISCRETION

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